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OAK RIDGE NATIONAL LABORATORY
Operated by
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SUBJECT: RADIOACTIVE WASTE MANAGEMENT AT OAK RIDGE NATIONAL LABORATORY

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FROM: F. N. Browder

ABSTRACT

The collection, treatment, disposal, and monitoring of radioactive wastes (solid, liquid, and gaseous) at Oak Ridge National Laboratory are described in detail. Illustrations of facilities, maps, and tables of data on waste volumes and radionuclides discharged to the environment are included. The philosophy and history of waste management are discussed. The report constitutes an evaluation of waste management at ORNL, concluding that the low degree of radioactive contamination of the air and water by ORNL does not represent a hazard to the local environment or population.

This report was prepared for the public hearings on industrial radioactive waste disposal before the Joint Committee on Atomic Energy, Congress of the United States, January 28 through February 3, 1959.

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RADIOACTIVE WASTE MANAGEMENT

At

OAK RIDGE NATIONAL LABORATORY

I. Introduction

A. General Description

Oak Ridge National Laboratory (ORNL) is a nuclear research and development center operated for the U. S. Atomic Energy Commission by Union Carbide Nuclear Company. The Laboratory employs a total of 4,000 people at two sites, X-10 and Y-12, near Oak Ridge, Tennessee. This report describes the management of wastes from the major radioactive operations at the X-10 site.

The main X-10 area is approximately one mile long by one-half mile wide. Its numerous buildings house laboratories, chemical processing pilot plants, radioisotope production plants, nuclear reactors, and supporting service facilities. Waste collection facilities are located near the center of the main area, and a waste disposal area is situated approximately one mile west of the main area. Figure 1 shows the arrangement of the ORNL X-10 site.

Radioactive wastes are collected and treated according to their physical states (solids, liquids, and gases). To simplify the removal of radioactive components before the wastes are discharged to the environment, the wastes are further classified into the following types for separate collection and treatment:

1. "Highly radioactive" liquid chemical waste: 7,000 gallons per day, 0.001 - 0.02 curie* per gallon, the "hottest" waste at ORNL, which is not highly radioactive compared to high-level wastes at nuclear production plants;

2. Liquid uranium waste (very small volume, moderate radioactivity);

3. Mildly contaminated process waste water - 700,000 gallons per day, 0.1 to 1.0 microcurie* per gallon;

4. Solid waste (variable radioactivity);

5. Reactor cooling air (high radioactivity);

6. Cell and hood ventilation air (low radioactivity);

7. High-radioactivity off-gas.

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*The curie is that quantity of any radioactive nuclide which disintegrates at the rate of 3.7×10^{10} disintegrations per second. This is the unit of radioactive substance equivalent to 1 gram of radium. A millicurie is 0.001 curie; a microcurie is 1×10^{-6} curie.

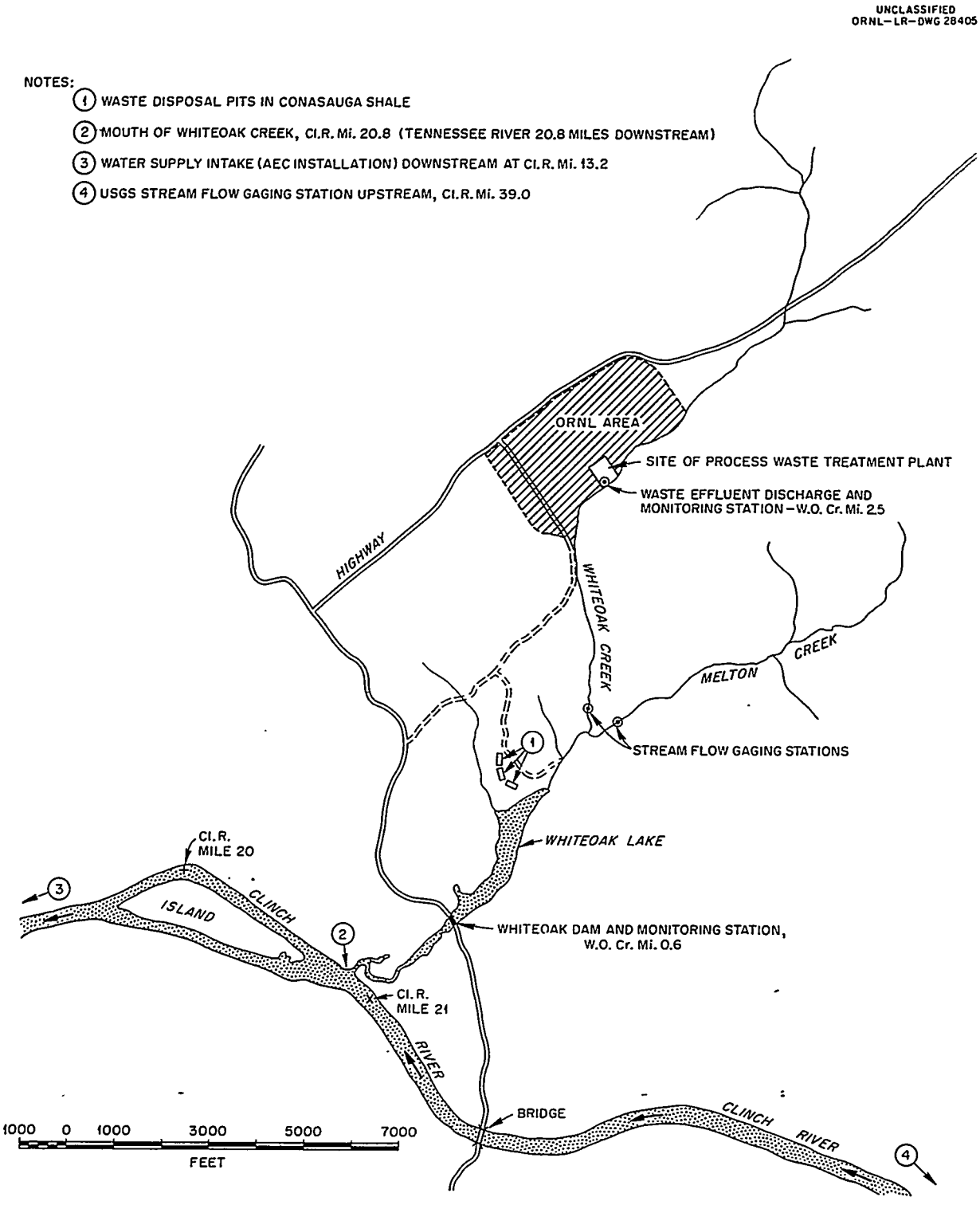


Fig. 1. Location Sketch Map, ORNL Area Surface Drainage.

These wastes are described in greater detail below. Figure 2 diagrams the management of these wastes (according to the above number sequence) and shows that after treatment to reduce the levels of radioactivity, liquid wastes are discharged to the Clinch River and gaseous wastes are discharged through tall stacks to the atmosphere. Solid wastes are buried in the ground.

B. Waste Management Philosophy

The goal of radioactive waste management at Oak Ridge National Laboratory is to dispose of the wastes as safely and as economically as possible. At present the following approach to this goal is being taken:

1. To confine the major portion of long-lived fission products from all types of waste in tanks or in the ground, where they will become chemically attached to the soil;

2. To dilute low-level wastes in the atmosphere and the surface water drainage system. The dilution factors available in these media are used to reduce the concentration of long-lived strontium-90 and cesium-137 to the maximum permissible concentrations set by the National Committee on Radiation Protection and the International Commission on Radiological Protection;[1]

3. To monitor the waste streams before and after discharge in order to follow the success of this program.

The disposal of radioactive waste into the ground is considered to be an experiment as well as an expedient solution to a local problem.[2] A number of local conditions permit this ground disposal experiment to be run with apparent safety:

First, the Laboratory is isolated by distance (12 miles to the nearest community) and by patrols that keep the public from potentially hazardous areas. A built-in factor of safety for the public is achieved by the fact that the nearest downstream consumer of Clinch River water is another nuclear plant (the Oak Ridge Gaseous Diffusion Plant--Item 3 in Fig. 1) in the Oak Ridge Control Area, which continuously monitors its water supply for its own protection.

Second, the levels of radioactivity and the volumes of waste from the research and development operations at the Laboratory are lower by at least 1,000 times than those at nuclear production sites such as Hanford and Savannah River.

Third, in the waste disposal area there is a large formation of Conasauga shale containing clays which have a considerable ability to capture and hold radioactive components of waste while allowing the waste water to seep through. The shale provides an inexpensive and effective means of disposal for most of the Laboratory's liquid and solid radioactive wastes. The shale pits are described in Section II, A. Both surface and ground waters from the main X-10 area and the waste disposal area flow to a natural basin drained by White Oak Creek, which empties into the Clinch River within the Oak Ridge Control Area. This restricted drainage provides a means of studying the lateral movement of any radioactive seepage from the waste disposal area.

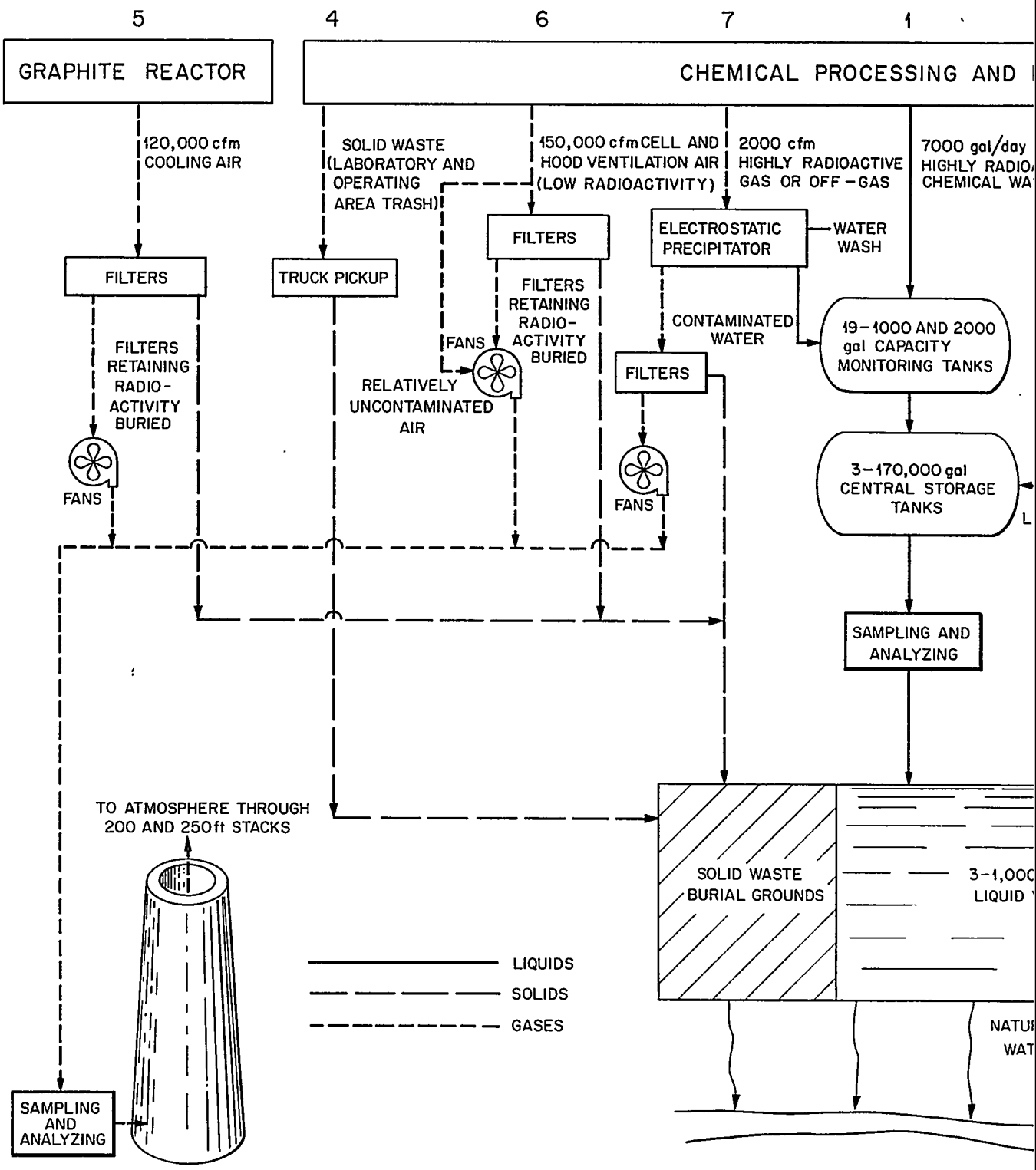
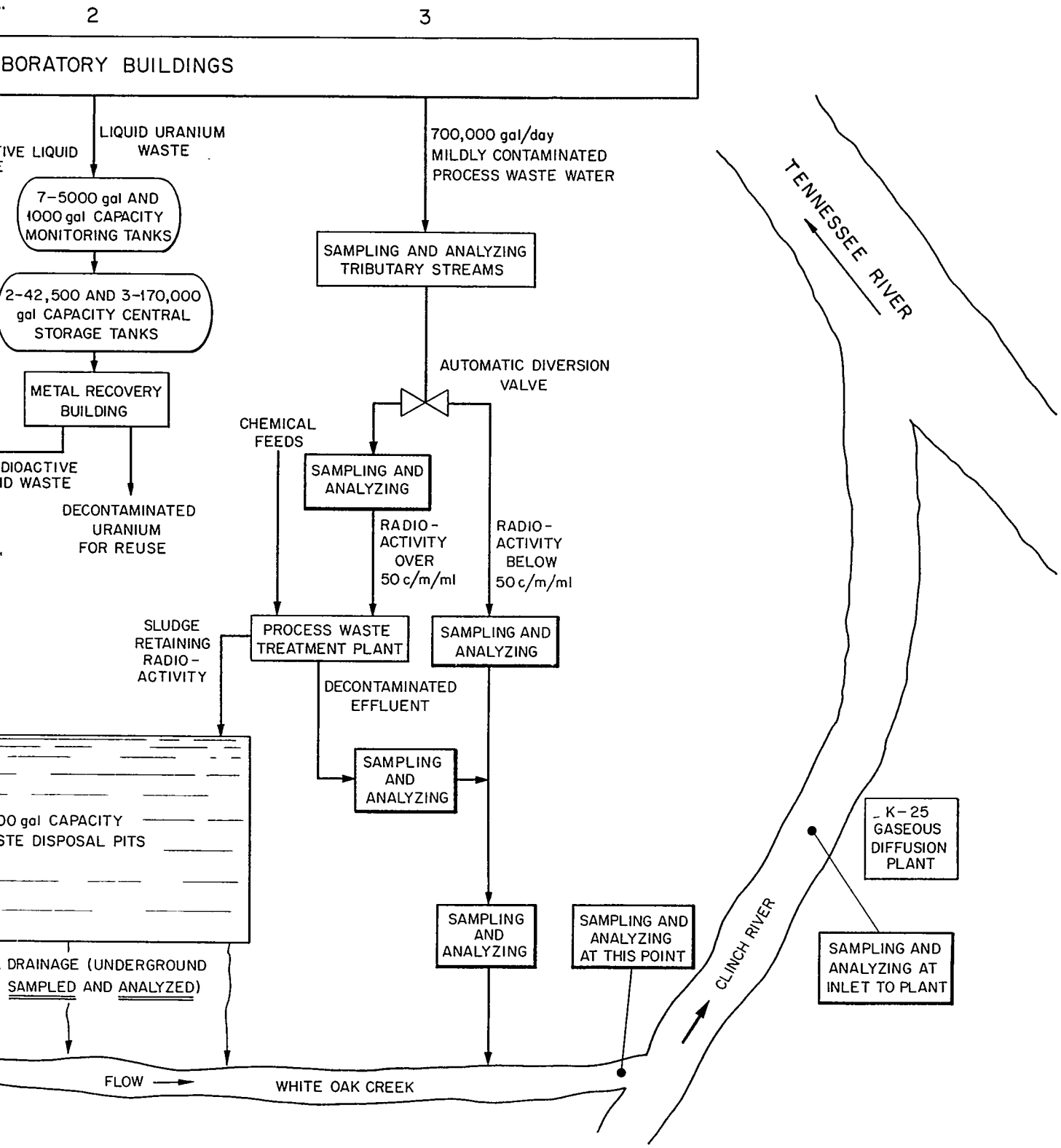


Fig. 2. Simplified ORNL



adioactive Waste Disposal Flowsheet.

One of the unique factors in dealing with radioactivity is that it cannot be destroyed. No process has yet been developed to speed up or slow down the emanation of particles and rays from radioactive atoms; time is the only agency capable of destroying radioactivity. This fact presents the large quantity producer of radioactive waste with two alternatives for disposing of it:

1. Rendering the radioactive waste harmless by diluting it and disbursing it until the concentration of the radioactive components is too low to be hazardous; or

2. Separating the radioactive components from the waste, concentrating them to a convenient volume, and storing them until time can destroy them by natural decay.

The waste management philosophy at Oak Ridge National Laboratory has been to follow the first alternative for large volumes of low-activity waste and the second alternative for higher-activity wastes.

Responsibility for segregating, collecting, treating, and disposing of all radioactive wastes since the beginning of the Laboratory has been delegated to the Operations Division, which must also measure and record all radioactive contamination as it is discharged to the environment. Responsibility for studying the effectiveness of radioactive waste disposal in the environment by means of stream sampling and surveys and responsibility for reporting the degree of contamination in these streams have been delegated to the Health Physics Division. The separate delegation of these responsibilities provides an independent check on methods of operation and analysis and on concentrations being released to the environment. Research and development efforts on waste treatment and disposal are carried cooperatively by the Chemical Technology and Health Physics Divisions.

C. History of Waste Management at ORNL[3]

The present size of the Laboratory and the scope of its work have been reached by a series of expansions. The means of handling the considerable quantities of waste produced by the processes performed at the Laboratory have had to be changed from time to time to keep pace with the expanding program.

The Laboratory was established in 1943 as a temporary pilot model for the Hanford, Washington, works. The Graphite Reactor, a chemical separations plant (the "Hot Pilot Plant"), and a number of large underground concrete (gunnite) tanks were constructed then. The tanks were intended to store all the highly radioactive liquid chemical waste and the liquid uranium waste accumulated during the life of the Laboratory, which was expected to be one year. However, expansion of the scope of the work in 1943 and indefinite continuation of the Laboratory increased the quantities of waste, necessitating a method of disposal to augment storage in tanks. It was decided to precipitate as much of the radioisotopes as possible in the storage tanks and to decant from the tanks those remaining in solution, dilute them with the Laboratory's large volume of process waste water, and disperse them into White Oak Creek. A portion of the precipitated radioisotopes remain as a sludge in the storage tanks at the present time. A dam was built across White Oak Creek 1.7 miles below the Laboratory (Fig. 1) in the autumn of 1943 to create a controlled area for the discharge of radioactive waste. A settling basin of 1,500,000 gallons capacity was completed in July, 1944, to serve as the waste collection and sampling facility and as a stilling pond to permit radioactive solids to settle from the waste before discharge to the creek.

Additional decontamination of the radioactive supernatant by decay was gained by receiving and holding waste in one of the large storage tanks for as long a time as possible while decanting to the settling basin from another tank containing aged waste. This procedure allowed sufficient time for much short-lived (and hence more intense) radioactivity to decay before the waste was discharged to the creek. The isotopes removed by this procedure were 8-day iodine-131, 28-day cerium-141, 33-hour cerium-143, 41-day ruthenium-103, 12.8-day barium-140, and 40-hour lanthanum-140.

Thus the 7,000 gallons per day of highest activity waste at ORNL was given a precipitation step, about one month holdup for decay, triple settling (in the tanks, the settling basin, and in the lake behind the dam), and about 500,000 to 1 average dilution in the Clinch River. It was calculated at that time that a maximum 5 curies per day of mixed fission products could be discharged safely into the lake, and for several years this criterion was used. An average activity discharge of 284 curies per year, considerably below this level, has been maintained to date (see Table I). This method of disposal by dilution and discharge through a controlled natural drainage basin was continued until June, 1949. This method of disposal had been considered adequate as a temporary measure, but the agreement upon more stringent tolerances necessitated an improvement.

From June, 1949, until June, 1954, the highly active liquid chemical waste was concentrated by evaporation in a pot-type evaporator instead of being decanted and diluted in the lake. During this period the evaporator processed a total of 11,650,000 gallons[4], reducing this volume to 432,000 gallons of radioactive concentrate that were stored in the concrete tanks. The water boiled off from this waste contained an average of only 0.014% of the radioactive contamination entering the evaporator. The effectiveness of the evaporator is demonstrated by the fact that during the period of its operation only 14.5 curies per year came from the evaporator, although an average of 320 beta curies per year was discharged to the creek from the Laboratory. The remaining contamination came from process waste water and from accidental discharges (mainly leaking waste pipes and valves). The evaporator was taken out of service in June, 1954, after the first 1,000,000-gallon experimental ground disposal pit had been in operation for two years. Since that time the pits have received all the highly radioactive liquid chemical waste.

In 1950 the Laboratory was again greatly expanded in size and scope of operation. The waste systems were expanded to handle the increased waste volumes and levels of radioactivity. Monitoring systems for all three liquid waste types were devised to aid the collection and segregation of liquid waste. Underground stainless steel tanks to monitor separately the highly active chemical waste and the liquid uranium waste were installed near each building or area that is a source of either type waste. The tanks permit sampling and measurement of waste volumes and rates of accumulation from each source.

Table I

YEARLY VOLUMES AND RADIOACTIVITY OF DISCHARGES OF LIQUID WASTES TO WHITE OAK CREEK AT ORNL, 1950-1957[a]

Year	Total Volume, 10^6 gal	Gross Beta Activity, curies			Per Cent of Gross Beta Activity Identified with Specific Radionuclides									
		Settling Basin	Retention Pond	Total	TRE (-Ce)[b]	Ce	Ru	Zr	Cs	I	Sr	Nb	Ba	Co
1950	226.4	172	15	187										
1951	297.6	169	3	172										
1952	268.2	411	87	498										
1953	239.4	289	140	429	53.1	2.6	0.8	2.6	8.5	0.2	27.6	1.0	0.8	---
1954	164.3	237	17	254	34.6	19.1	0.5	1.0	20.3	0.5	23.3	0.4	0.3	---
1955	210.6	213	54	267	30.3	14.7	3.1	0.6	31.6	0.2	18.7	0.6	0.2	---
1956	260.7	253	20	273	24.4	12.3	2.0	0.5	42.1	0.1	15.1	1.0	0.2	2.2
1957	272.3	(combined)		189	25.8	4.4	1.0	10.2	36.5	0.0	18.0	0.7	0.0	1.8

a. Volume of settling basin effluent measured in weir box with liquid level float recorder. For determination of radioactivity, continuous proportional samples of the effluent are composited for daily gross beta measurements and monthly radiochemical separations and analyses.

b. Trivalent rare earths exclusive of cerium.

The monitoring scheme installed for process waste water consists of weirs mounted in manholes in the underground sewer system that collects this waste. These weirs permit measurement of the waste volume from each source and proportional sampling of the waste for determination of the radioactive contamination.

For gaseous wastes a 250-foot stack, filter houses, blowers, underground ducts, and an electrostatic precipitator were added to existing facilities.

In 1951 the first experimental ground disposal pit was built. A larger (1,000,000 gallons capacity) pit was built in 1952, and two more 1,000,000-gallon pits were built in 1955. A pump and a 1-1/2-mile-long pipeline from the collection tank area to the disposal pit area were installed in 1954 to replace a tank truck previously used to transport liquid waste.

Between 1952 and 1957 a metal recovery process reclaimed approximately 130 tons of uranium from liquid uranium waste collected over the years in the original waste storage tanks. The waste from this recovery process was sent to the highly radioactive liquid waste system for evaporation or disposal to pits.

The lake behind the dam on White Oak Creek was drained in 1955 for the following reasons:

1. To perform necessary maintenance work on the dam facilities;
2. To destroy and dispose of the aquatic species in the lake;
3. To avoid attracting and harboring migratory wildfowl;
4. To provide additional safeguards by increasing retention potential;
5. To facilitate and improve control of activity releases;
6. To permit modification of sections of the lake area for research use.

The stream now flows through the lake bed and through a sluice in the dam, which can be closed to impound contaminated water when the need arises. A bypass has been proposed to carry the stream flow around the dam when a need arises to impound water, but this proposal has not been carried out to date. A continuous sampler and a radiation monitor (submerged in a container through which stream water is circulated) have been installed at the dam. The monitor can detect a "slug" of radioactivity and can sound an alarm, but it is considered a stop-gap instrument until a better one can be developed.

Studies of wildlife in the lake were made between 1950 and 1953[5] and in 1956[6,7] to determine some ecological effects of radioactive contamination. Agricultural crops are being grown in the contaminated mud of the lake bed to study uptake of radioactivity by plants[8,9]

In 1957 a waste water treatment plant was completed and put into operation. The function of this plant is to reduce the level of radioactive contamination in the low-activity process waste water discharged to White Oak Creek. An automatic diversion valve is currently (January, 1959) operating to feed the

entire process waste water flow to the treatment plant whenever the level of radioactivity in the waste exceeds a given point. When the activity level is below the set point, the automatic valve diverts the waste around the treatment plant and to the creek through the settling basin.

A multicurie fission product pilot plant was completed in 1957 to recover strontium-90, cesium-137, and other valuable radioisotopes from high-activity liquid wastes. This pilot plant should be most valuable for treating extremely high-level waste from future processes planned for the Laboratory.

II. Current Handling and Disposal Methods for Liquid Wastes

A. Stream No. 1 (Fig. 2): "Highly Radioactive" Liquid Chemical Waste

The term "highly radioactive" is applied to the 7,000-gallon per day liquid chemical waste stream at Oak Ridge National Laboratory only because this stream has the highest level of radioactivity of any waste at the Laboratory. However, compared to highly active wastes (containing 80 to 5200 curies per gallon[10]) at production sites, this ORNL stream is misnamed, because its concentration of radioactive components is normally between 0.001 and 0.02 curie per gallon. In 1957 some experimental operations (Thorex) produced wastes having activity levels as high as 2 curies per gallon when accepted into the waste system, but the total volume of these wastes was less than 1,000 gallons. Because the research and development nature of the Laboratory brings about frequent changes in the processes that produce waste, the waste composition is not consistent. Larger volumes and higher radioactivity levels are expected from future operations. The main radioisotopes are usually cesium-137, ruthenium-106--rhodium-106, strontium-90--yttrium-90, and trivalent rare earth elements. Strontium, cesium,and trivalent rare earths constitute the major fraction of radioactivity on an average disintegration per minute basis. Sodium and nitrate account for about 70% of the nonradioactive solids in the waste.

The handling of highly radioactive liquid chemical waste is illustrated in Figs. 2 and 3, which are simplified flowsheets of the Laboratory's waste systems. The main sources of the liquid chemical waste are chemical processing pilot plants. Radioisotope production facilities and research laboratories produce wastes of smaller volume and lower activity level. The special fission product wastes shown in the upper right corner of Fig. 3 are recovered and consequently do not contribute directly to the waste stream.

Highly radioactive liquid chemical waste is discharged from process vessels in laboratory and pilot plant cells into "hot" drains, which are stainless steel pipes leading to underground stainless steel monitoring tanks. There are now 19 of these 500 to 4000-gal capacity tanks in service (including tank W-1, Fig. 3); their total capacity is approximately 34,000 gal. Each tank is located near its main contributor to permit gravity flow to the tank. The function of these monitoring tanks is to collect the waste and to provide a means of sampling it and of measuring its volume. The tanks provide the Operations Division with a means of checking on waste contributors for rate of waste production and to be sure each type waste gets into its proper system for treatment before disposal. Each tank has connections to the highly radioactive chemical waste system and to the process waste water system to give a choice of treatment depending on composition, radioactivity level and other considerations.

Figure 4 shows the burial details of the monitoring tanks. Each tank is anchored to a concrete saucer that slopes toward a sump. The sump collects ground water seepage and any liquid that leaks from the tank. Liquid in the sump is sampled periodically for radioactivity through a "dry well" to determine whether the tank has leaked. During the past 15 years three tanks were emptied and abandoned after leaks were discovered in this manner. Because most of these leaks were caused by acid corrosion of the tanks, caustic soda to neutralize incoming acid waste is now added to each tank. No leaks have been discovered since this procedure was started.

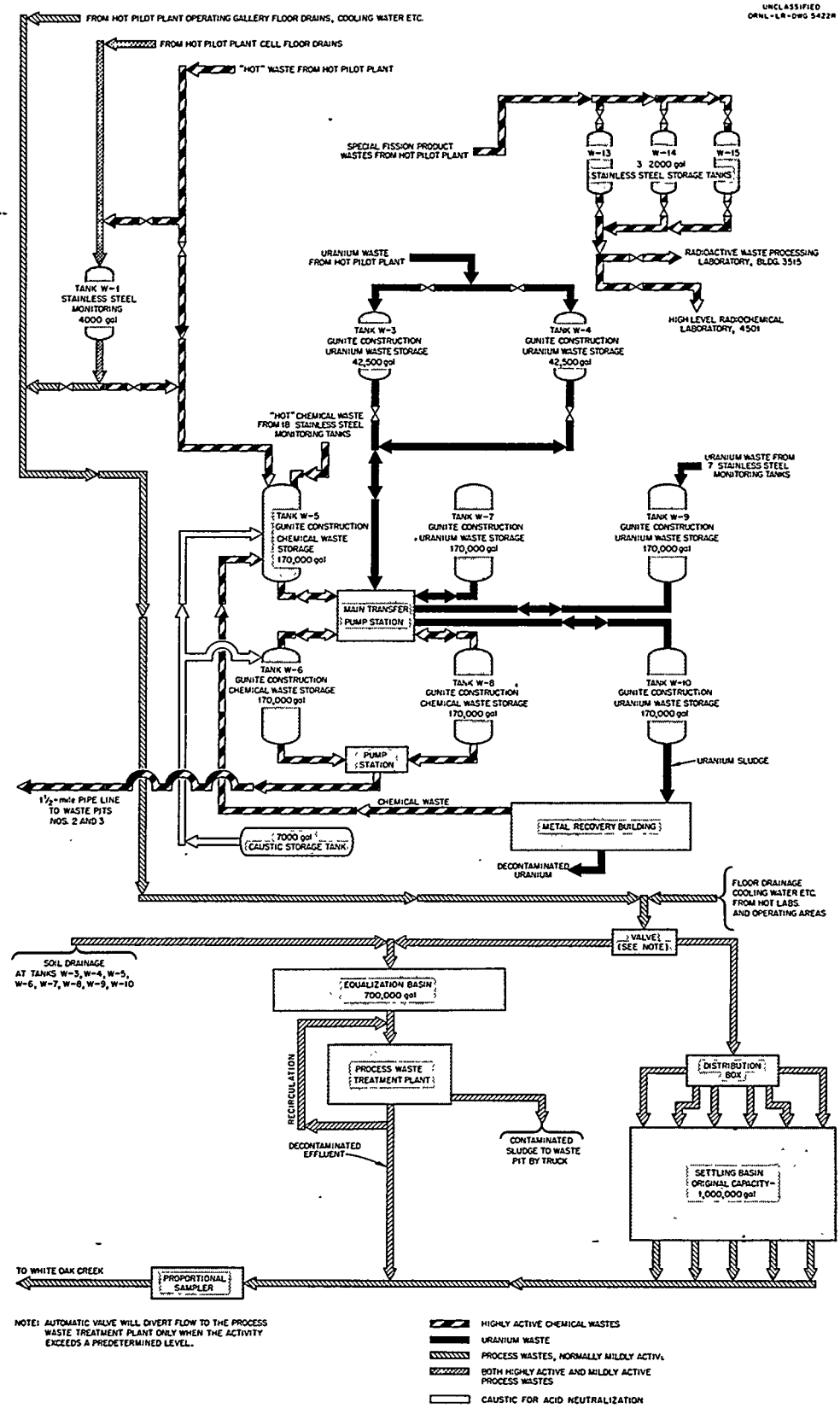


Fig. 3. ORNL Simplified Liquid Waste Disposal Flowsheet.

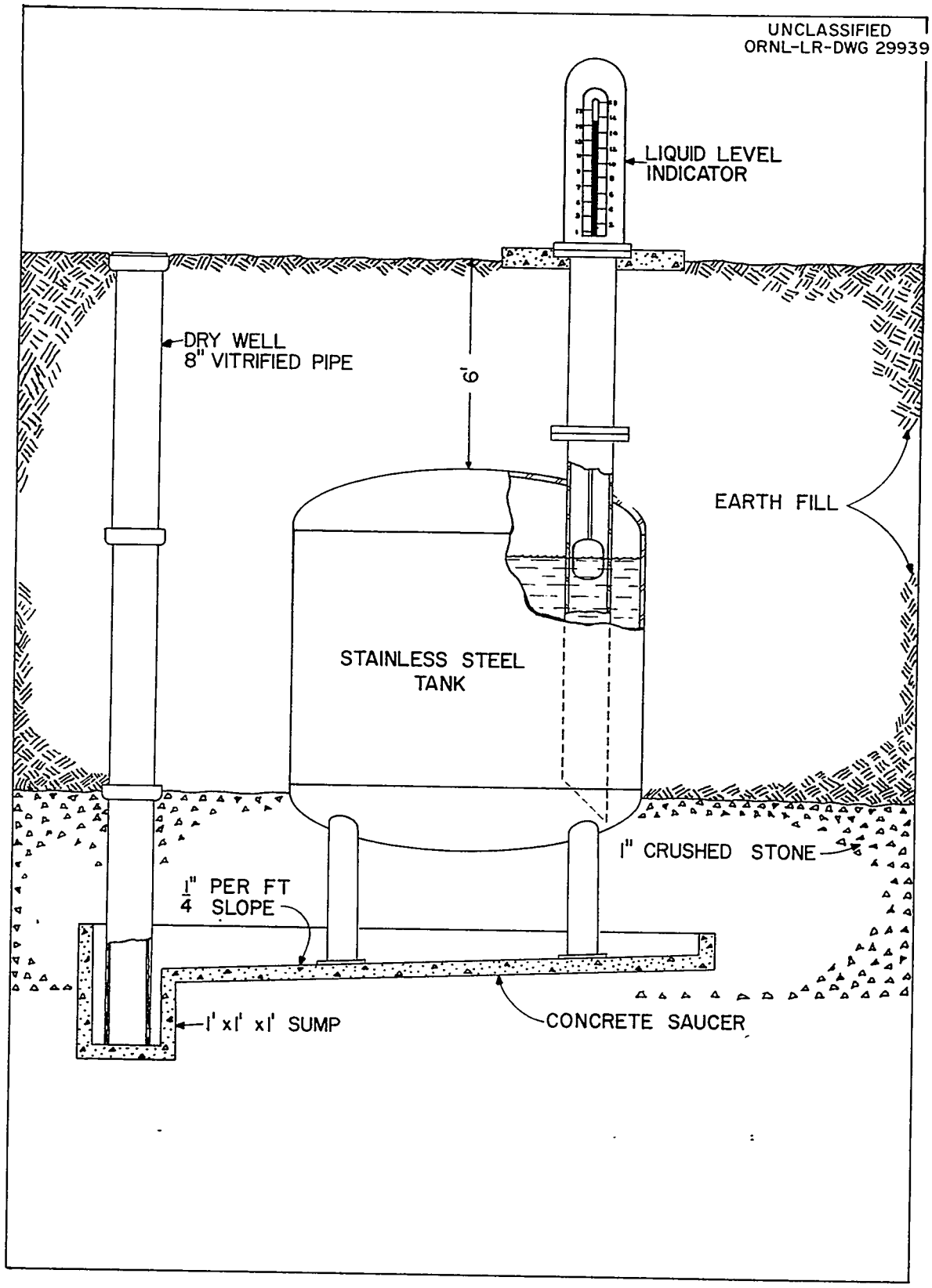


Fig. 4. Dry Well and Waste Collection Tank Installation.

Each tank is equipped with a float-type volume gauge, which shows on a board above ground the quantity of waste in the tank. Although the rate of waste accumulation in all 19 monitoring tanks is continuously recorded at a central station by a telemetering system, a waste system operator visits each tank every four hours to check the waste volumes lest a tank overflow because of a failure in the telemetering system.

When a monitoring tank becomes filled, the operator switches on a self-priming, packless pump to transfer the waste to one of the three 170,000-gallon concrete storage tanks in the storage farm centrally located in the main X-10 operating area. These tanks rest on concrete saucers that drain to a system of dry wells for leak detection similar to those of the monitoring tanks. The central storage tanks also have volume gauges, samplers and an underground system of transfer lines that permit transfer of the wastes from one tank to another.

The function of the central storage tanks is to provide temporary storage for the highly radioactive liquid chemical waste while the short-lived radioisotopes decay (see section I, C). These tanks provide adequate surge capacity to accommodate normal accumulation from Laboratory operations even during occasional shutdown and repair of pumping equipment. Wastes are periodically pumped from the central storage tanks approximately a mile and a half through a 2-in. dia underground steel pipe to the disposal pits.

The waste pits are three 1,000,000-gal open cavities bulldozed in the earth in a location chosen for remoteness from the Laboratory, the type soil, and the fact that underground drainage is toward White Oak Creek.[2,11a] The soil, Conasauga shale, has the property of removing and retaining most of the radioactive components while the waste water and certain nonradioactive chemicals seep through it toward White Oak Creek. Waste enters pit No. 3, which overflows through a valved pipe to pit No. 2, which overflows similarly to pit No. 4. Figure 5 is an aerial photograph of the waste disposal area showing the pits in the foreground, the creek, and in the upper left corner a part of the main X-10 area. The pits are each 15 ft deep with sides sloping at an angle of 30°. Their top dimensions are 210 ft by 100 ft. The pits are covered with wire screen to prevent access to wild life. The waste discharged into the pits is sampled and analyzed for radioisotopes and stable chemical ions, and the movement of these materials in the soil and in the seepage into the creek is monitored by the Health Physics Division. The only radioisotopes detected in the seepage to date are ruthenium-106, cobalt-60, and antimony-125. By the end of 1956 the seepage of ruthenium-106 was estimated to average 23 curies per year. In 1957 samples indicated 200 curies of ruthenium-106 seeped from the pits, but only 60 curies of ruthenium-106 were detected passing White Oak Dam. In 1957 a total of 41,900 curies was discharged to the pits, an increase of 20% over 1956; in 1958 52,800 curies were discharged, an increase of 25% over 1957. These increases were due mainly to higher radiation levels in processing at the Hot Pilot Plant. By the end of 1958 the total of all wastes discharged to the pits since the start of this practice was 11,703,000 gallons containing 167,000 curies (at time of discharge).

B. Stream No. 2 (Fig. 2); Liquid Uranium Waste

The quantity of liquid uranium waste is much less than it was earlier in the history of the Laboratory, as it is now produced at a maximum rate of only 100 gallons per week. In the sense that it is actually a solution of reusable uranium

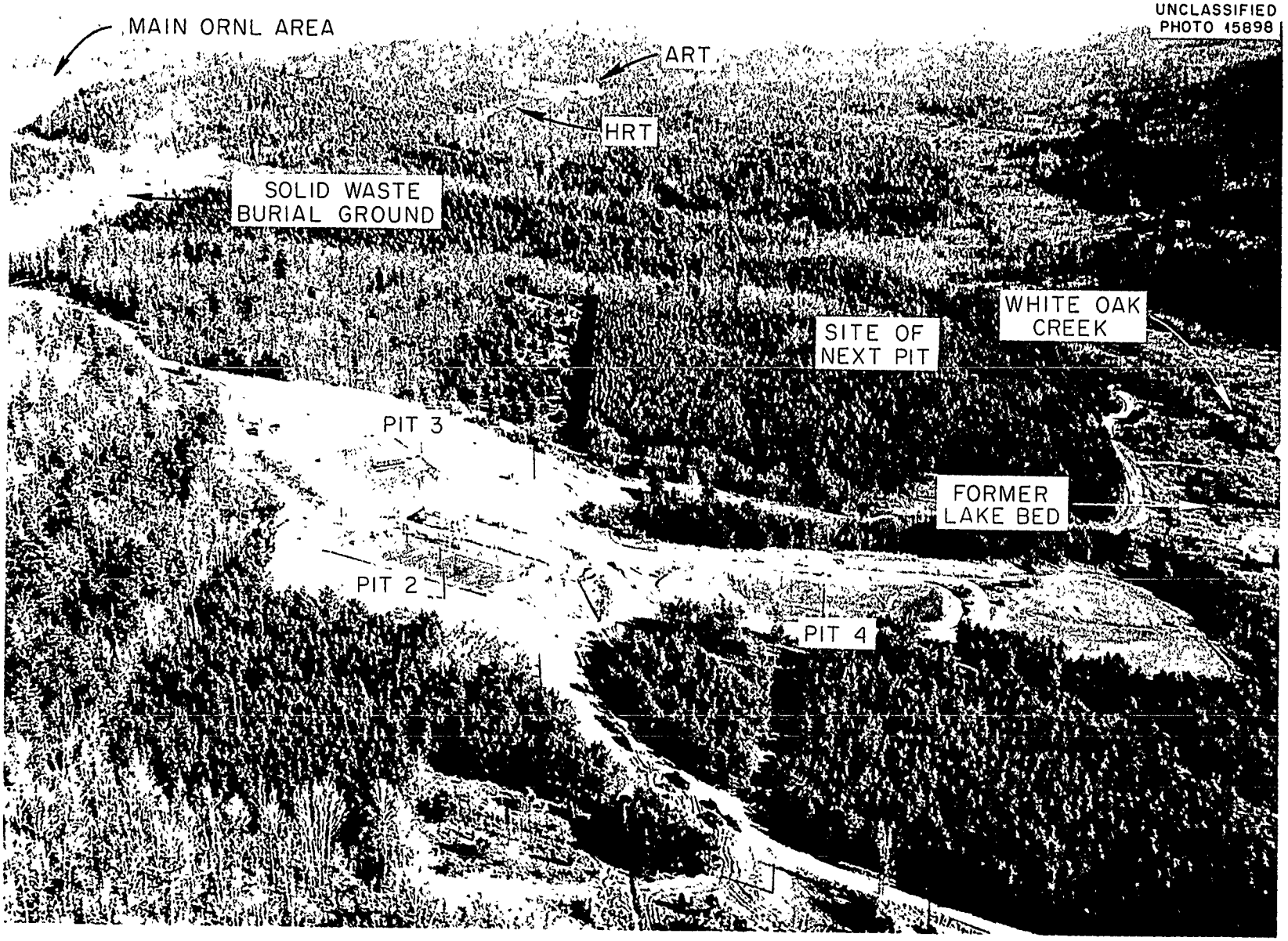


Fig. 5. ORNL Waste Disposal Area.

contaminated with fission products, this is not a true waste. It is collected and stored temporarily prior to treatment for recovery of the uranium. Between 1952 and 1957 approximately 130 tons of uranium were recovered. The separated fission products are discharged to the highly radioactive chemical waste system. Figures 2 and 3 illustrate the handling of liquid uranium waste.

The liquid uranium waste from the various contributors is collected separately from other liquid waste by means of a system of seven monitoring tanks (total capacity 7,500 gallons), similar in all respects to those described above for the highly radioactive liquid chemical waste. Each buried tank serves a source of liquid uranium waste, which flows by gravity through special stainless steel drains from the source facilities to the tanks. The waste volumes in the tanks are recorded at the central station by the previously described telemetering system, and a waste system operator checks the tank gauges every four hours to prevent overflows. The waste from full tanks is pumped to the central tank farm, where three 170,000-gallon and two 42,500-gallon underground concrete tanks store it until enough has accumulated to warrant running the uranium recovery facility.

C. Stream No. 3 (Fig. 2): Process Waste Water

The sources of process waste water are equipment cooling systems, floor drains, decontamination pad drains, storage canals, laboratory sinks, and discharges from low activity operations. It is the least radioactive of all Laboratory liquid wastes except sewage and storm water, yet it is the most difficult to manage because of its combination of radioactivity and large volume. The 700,000 gallons per day of this waste make storage impractical and necessitate disposal on a current basis. The waste is collected, sampled and discharged to the creek continuously. Figures 2 and 3 illustrate the relation of the process waste water system to the other liquid waste systems.

The process waste system sometimes serves as an emergency "warm" system. Much of the radioactive contamination put through this system is a result of equipment failure, human error, or accidents that cause a misdirection of contamination from the highly active chemical waste system. Whenever unusually high levels of radioactivity occur, an effort is made to divert the radioactive portion to the high activity level system as soon as possible. A network of 6-in. to 30-in.-diameter vitrified clay pipes collects and conveys the process waste water by gravity flow to a central monitoring point near the inlet to the 1,500,000-gallon settling basin, where the volume of flow is measured and sampled continuously. The samples are collected every four hours and analyzed for gross beta activity. The process waste water collection system is divided into several sections, each of which is served by a strategically located monitoring station. Each station is a concrete manhole in which are mounted a V-notch weir, a water level recorder for determining volume of flow and a finger-type pump for collecting samples whose volumes are proportional to the flow. When the radioactivity level at the central monitoring station rises to about 150 c/m/ml* or greater, the source

*c/m/ml = counts per minute per milliliter. All results are referred to second-shelf counting using an end-window Geiger counter.

responsible for the increase can be located by referring to the monitoring samples, and corrective action can be taken to minimize discharges of radioisotopes through this low-level system. With a new treatment plant available, process waste water can be temporarily held and treated to reduce the level of activity. The treatment plant is described in more detail below.

The yearly volumes and radioactivity of the process waste water during the years 1950 to 1957 are summarized in Table I. The total volume of flow has ranged from about 165 to 298 million gallons per year for an average of 700,000 gallons per day. The level of gross beta activity normally ranges from less than fifty to several hundred c/m/ml with occasional transient levels above 1000 c/m/ml, depending upon operating conditions in the Laboratory. The fourth column in Table I, headed "Retention Pond," refers to a small volume of drainage from the monitoring pads underneath and from the soil around the underground waste storage tanks in the central tank farm. This pond previously served as a monitoring point for the detection and measurement of any leakage in the highly radioactive liquid chemical and liquid uranium waste piping and storage systems. The retention pond stream has been intercepted and is now pumped to the equalization basin of the new process waste treatment plant. The total gross beta activity of discharges to White Oak Creek from the process waste water system and the retention pond has ranged from 172 to 498 curies per year. Despite a hundred-fold increase in levels of radioactivity in the operations at ORNL since 1951, Table I shows that the activity levels in discharges to the creek have not increased proportionally. The 1957 low of 189 curies was achieved by a reduction in the frequency of accidental releases, constant policing of process waste streams, and operation of the process waste water treatment plant. Less than 100 curies were discharged in 1958. The waste discharged from the collection system is normally clear with very little suspended matter; but it varies widely in acidity or alkalinity, the pH ranging from about 2.0 to 11.5.

The flow of the effluent from the settling basin is measured at a weir box near the point of discharge to the creek, and composite samples are collected by a Trebler proportional sampler. Radiochemical analyses of these samples for five years (1953-1957) are summarized in Table I. The chemical, radioactive, and physical properties of the waste discharged from the settling basin may be influenced by sedimentation in the basin or by heavy growths of algae, both of which are capable of concentrating radioactivity.

Process Waste Water Treatment Plant[12, 13, 14]

Increases in the chemical processing operations and in their radioactivity levels at ORNL during several recent years increased the volume and activity level of process waste water discharged to the creek, as shown in Table I. On several occasions surveys revealed that the level of radioactivity in the Clinch River for short periods exceeded the recommended average concentration limit for unidentified radionuclides.[1] These incidents emphasized the need for a treatment plant to reduce the level of radioactivity in the process waste water. The low levels of radioactive discharge in 1957 and 1958 demonstrate that the treatment plant operation plus monitoring efforts within the Laboratory have been successful.

The radioactive contamination discharged to the creek has always passed through the process waste water system as a very dilute solution chemically similar to "hard" water. The over-all results of extensive laboratory and pilot

plant studies indicated that the process waste water treatment plant should be designed to use a horizontal flow, lime-soda water softening process with provision for alternative use of phosphate coagulation to effect strontium removal and the addition of clay to increase the removal of cesium.

The plant, completed in August of 1957, is located near the central monitoring and diversion station at the outlet of the process waste water collection system (Fig. 1). It has a design capacity of 500,000 gallons per day, and provision has been made for future expansion to double this capacity. In addition to the routine treatment of waste, the plant is designed to give special treatment to waste containing abnormally high concentrations of radioisotopes caused by accidental or emergency releases.

Figure 6 is a diagram of the facilities for treatment and disposal of process waste water. When the inlet valving is operated manually, the plant receives all process waste water flowing between 8:00 a.m. and 8:00 p.m., the period when most of the contaminated discharges occur. This method results in (1) the treatment of large volumes which might not need decontamination and (2) the possible bypassing of active wastes at night. Therefore, an automatic diversion valve was provided which, in response to submerged Geiger-Mueller tubes, permits low level wastes to bypass the plant but diverts into the equalization basin all wastes which contain a greater concentration of radioactivity than a predetermined amount.[15] The 700,000-gal capacity equalization basin serves to minimize fluctuations in the composition of the plant influent and to supply waste to the treatment plant as needed.

Figure 7 is a cut-away view of the process waste water treatment plant. Uniform flow rate through the plant is maintained by two 350-gpm centrifugal pumps drawing from the equalization basin. Two gravimetric feeders apply slurries of lime and soda ash to the flash mixer which has a detention time of 1.5 minutes. (A third feeder is available for clay, trisodium phosphate, etc., as needed.) The three coagulation basins in series, providing 30 minutes of slow mixing, are followed by 2 hours settling in a 12 x 70 x 8 ft deep basin. The effluent is discharged to the creek. It can be recycled through the plant for additional treatment, although this has not yet been attempted to date.

The sludge that precipitates in the settling chamber contains the radioisotopes removed from the waste. A sludge scraper operating continuously moves it along the concrete bottom of the settling chamber into hoppers at the deep end of the chamber. Valves permit the sludge to drain from the hoppers to a 16,000-gallon concrete tank for short-time storage before disposal. Two plunger-type pumps lift the sludge to a partly shielded tank truck, which transports it to the disposal pits for highly radioactive chemical waste.[2] A part of the sludge can be recycled through the treatment plant when this is desirable.

The mechanical equipment is designed to allow maintenance without draining the chambers, thus utilizing the waste water as a shield to protect personnel from exposure to radiation from the radioactive sediments in the chambers.

During the first 11 months of operation 53 to 87% of the gross beta activity was removed by a single pass of the waste through the treatment process. The removal of strontium-90, considered to be the most hazardous component, ranged

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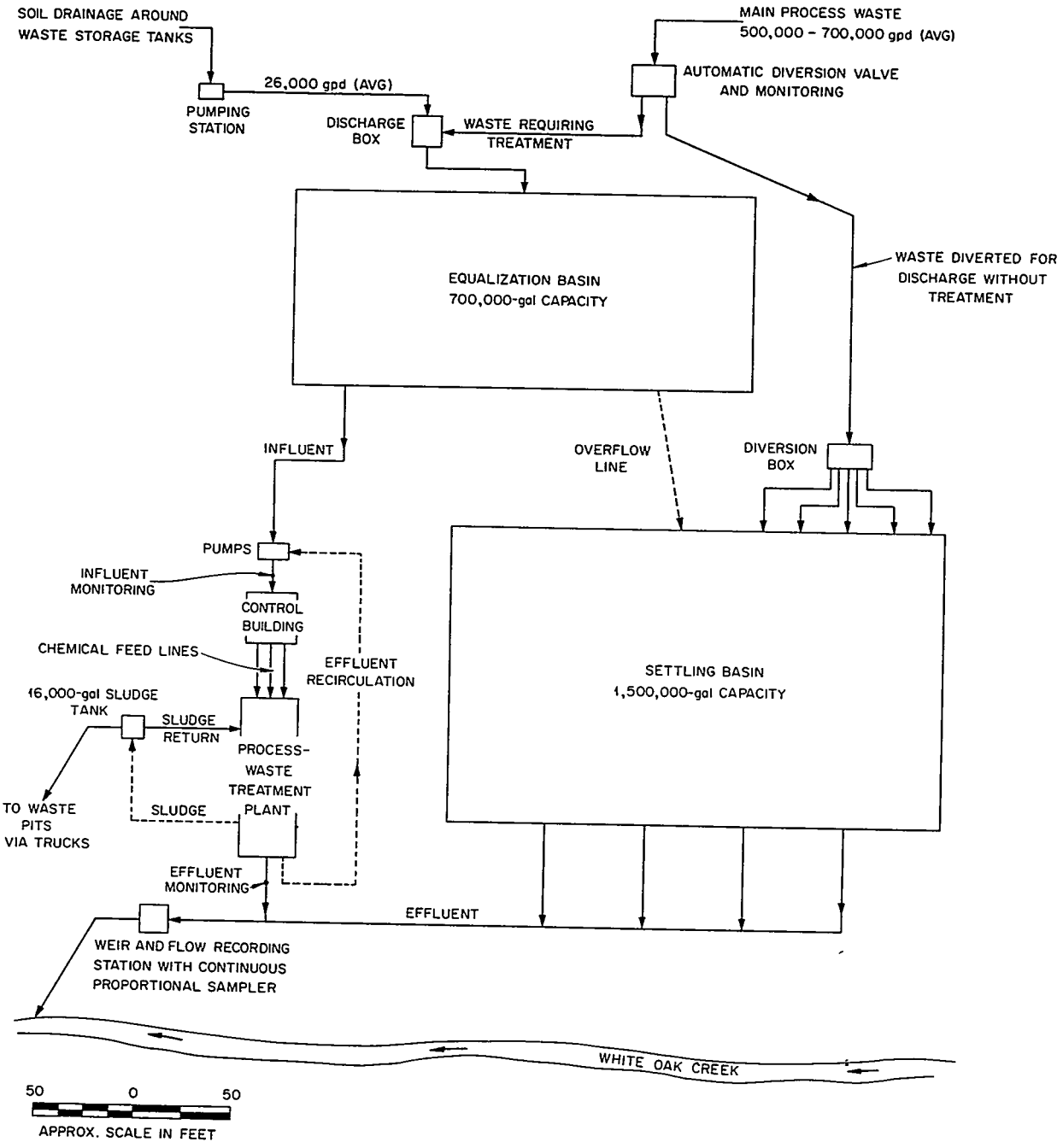

Fig. 6. Flow Diagram of Process Waste System.

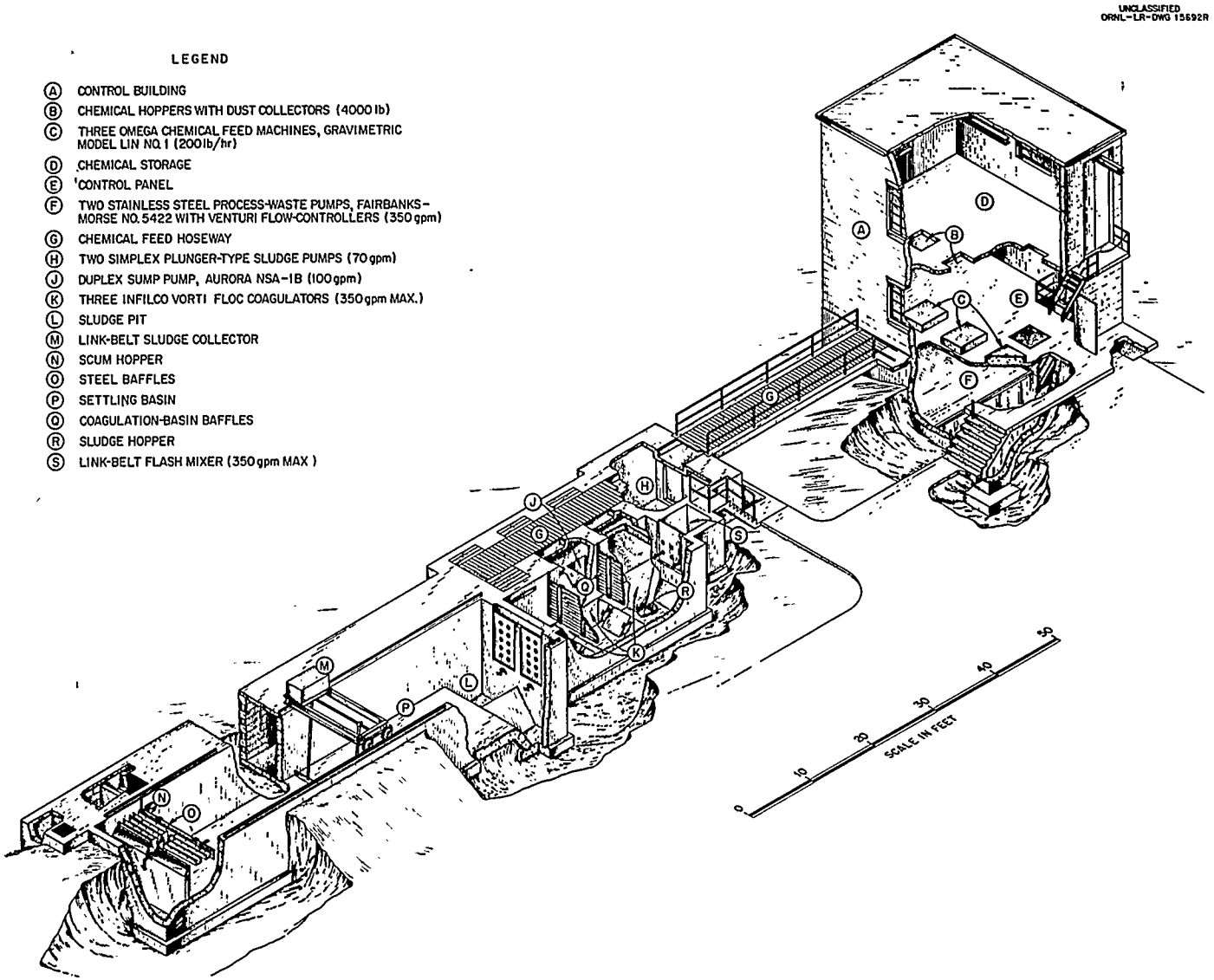

Fig. 7. Process Waste Treatment Plant.

from 66 to 87% for monthly periods and has been as high as 94% during one week. The trivalent rare earth elements were removed as efficiently as the strontium; about 17 curies of strontium and 15 curies of total rare earths were removed from the process waste during this period. Cesium removal has been low because clay has not been added to the process on a continuous basis, but during one week 30 ppm of a local clay increased cesium removal from 16% to 55%. (A commercial source of a similar clay suitable for continuous use has been located.) The operating experience to date indicates that this facility will fulfill the primary purpose of increasing the safety factor in control of radioactive waste released to the Clinch River.

D. Wastes Discharged Directly to Creeks

There are several comparatively minor sources of radioactive waste not included in the main ORNL liquid waste control systems listed in Fig. 2 because of remote location or other considerations. Normally these sources produce low activity waste, most of which is monitored before release from the source facility to the nearest creek. These sources are:

The Low Intensity Test Reactor (LITR). Two retention ponds catch the primary coolant (water) when the reactor is drained or when it leaks coolant. Cooling water from irradiation experiments in the reactor also is discharged to these ponds. Most of the radioactivity is caused by sodium-24, which decays rapidly. The ponds are periodically drained to a branch of White Oak Creek only after the water radioactivity has decayed to the extent that a scintillation counter reads lower than 100 gross beta and gamma counts per minute per milliliter. The LITR can discharge its higher activity waste to a monitoring tank in the highly radioactive chemical waste system previously described.

The Oak Ridge Research Reactor (ORR). The LITR retention ponds also catch reactor water drainage from the "warm" sump in the ORR building. Higher level wastes can be handled through the same monitoring tank that serves the LITR.

The Homogeneous Reactor Test (HRT). The HRT facilities are located in Melton Valley across a high ridge from the main X-10 area (Fig. 5). A retention pond, excavated in Conasauga shale near the reactor building receives reactor storage-canal water and cell shield water when these chambers are drained. The pond overflows to Melton Branch, a tributary of White Oak Creek (Fig. 1). The capacity of the pond equals the entire volume of all the HRT building cells, which are filled with water to shield personnel when repairs to equipment in the cells are required. The level of radioactivity in the water should normally be quite low. In the event the activity level is high enough to make discharge to the pond undesirable, the water can be drained to a 12,000-gallon stainless steel tank buried near the pond. After concentration by an evaporator this waste can be hauled in shielded carriers to the highly radioactive waste disposal pits previously described. The evaporator effluent can be recycled or discharged to the pond.

The Aircraft Reactor Test (ART). The ART facilities are located away from the main ORNL area and near the HRT site in Melton Valley. Although the facilities are not now in use, they can be used for radioactive operations. Drains in the reactor building lead to a sump from which low-level waste can be pumped to a retention pond. This pond is excavated in the shale near Melton Branch about 300 yards from the reactor building. No provision has been made to drain or to overflow the pond.

The Tower Shield Facility (TSF). The TSF, which contains two water-cooled reactors, is located near the Clinch River two miles from the main X-10 area. An open pool 20 ft square by 25 ft deep and an adjoining equipment storage basin 4 x 8 x 12 ft deep serve these reactors. The water circulates through one of the reactors and returns to the pool, which periodically overflows downhill away from the facility. A one-gallon sample is taken from the pool once a week for radiation analysis, and the mud on the hillside is sampled when overflows occur. Sodium-24 and irradiated water impurities comprise almost all the radioactive contamination in the water. Four 20,000-gallon tanks are available at the TSF site for contaminated water holdup in the event fission products get into the water from a ruptured fuel element.

The other reactor has a closed-circuit cooling system equipped with a water demineralizer. Most of the radioactivity will be caught and will decay in the demineralizer, which will be drained to the pool periodically.

The ORNL Laundry. The contaminated-clothing laundry discharges its waste water directly to White Oak Creek. Normally this waste is not monitored before discharge because the clothing is monitored to avoid washing of highly contaminated material and because contamination is diluted further by the washing process.

The ORNL Rolling Mill. The rolling mill, where uranium fuel elements are fabricated, has floor drains that discharge directly to a branch of White Oak Creek. Rolling mill waste is not normally monitored before reaching the creek.

Fallout from ORNL Gas Disposal Stacks. Radioactive particles from gas disposal operations can fall throughout the ORNL drainage area and thus get into the streams. Continuous monitors measure the fallout, and rain gauges at the air monitoring stations gather samples for radioactive analysis. The air monitoring program is described in section V, A.

Although the very dilute wastes discharged directly to the creeks, as described in this section, are not included in any of the main ORNL liquid waste control systems, monitoring of the creeks downstream affords a reasonable measure of control. The stream monitoring program is described in section V, B.

III. Current Handling and Disposal Methods for Gaseous Wastes

Air contaminated with radioisotopes constitutes the greatest volume of radioactive waste but not necessarily the greatest hazard created by the operation of nuclear reactors and associated chemical processes and laboratories. Basically, there are three general types of air-borne contaminants. The nature of the hazard involved and the technology required for its elimination differ for each type:

1. Radioactive noble gases, xenon, krypton, etc., which are chemically inert, do not constitute a serious hazard at ORNL and are controlled by releasing the contaminated air from tall stacks for dilution by the atmosphere. (400 curies per day of argon-41 are discharged).

2. Chemically reactive gases, such as I_2, are controlled by adsorption equipment (usually caustic scrubbers) located in the process areas.

3. Radioactive solid and liquid particles suspended in air are controlled effectively by filters and electrostatic precipitators.

All air that has a possibility of becoming contaminated is exhausted to the atmosphere from three tall stacks for dispersal. Standards for safe discharge are set and released waste gases are monitored by the ORNL Health Physics Division. The air passing through the stacks is continuously sampled and analyzed, and the fallout is continuously checked at 17 locations within seven miles of the Laboratory. One station (Berea, Kentucky) is about 120 miles away. The volumes of air and their flow patterns through the buildings are controlled to minimize the amount that has a possibility of becoming contaminated. Contamination of large quantities of air surrounding process equipment is avoided by designing the equipment to be airtight and by operating all process under a negative pressure. The air-borne radioactivity released during processing is thus caught in a relatively small amount of air, called "vessel off-gas", which is treated before release. The air surrounding the vessel in concrete cells, designated "cell ventilation gas," thus has a small possibility of becoming contaminated and is handled on a different basis from the vessel off-gas. These methods of handling the large quantities of air passing through the Laboratory facilities sharply reduces the amount of air requiring treatment before discharge.

The types of air handled at ORNL and whether or not each is treated before discharge can be briefly summarized as follows:

Office Ventilation: Six changes of air per hour, without treatment;

Laboratory Ventilation: Minimum of ten changes of air per hour, without treatment;

Laboratory Hood Ventilation: Minimum of 50 feet per minute face velocity for adequately designed hoods, with provisions for the installation of a filter in the discharge duct when necessary. In addition, each hood is equipped with two vacuum systems, the first system with 10 inches of water vacuum to draw gases from vessels containing high levels of radioactivity and the second system with 20 inches of mercury vacuum to be used for solution transfers and other

high vacuum applications. The air from both these vacuum systems is collected and treated as vessel off-gas before being discharged to the atmosphere.

Cell Ventilation: Minimum possible air flow; each cell is held at a reduced pressure (up to 1 in. of water vacuum) so that air leakage will be inward. The air drawn from cells where there is a high risk of its becoming contaminated (usually cells containing pilot plant or experimental equipment) is filtered. All cell ventilation air is discharged from the tall stacks for dispersal to the atmosphere.

Vessel Off-gas: Minimum possible air flow in a system with a vacuum of 40 inches of water for dissolver and process vessel off-gas lines. The air from this vacuum system is treated before discharge.

Reactor Cooling Air: All air used in cooling a reactor is filtered.

Air Cleaning Facilities

A central gas disposal facility has been established at Oak Ridge National Laboratory to clean the radioactive contaminated air from the chemical processing areas and to dispose of it to the atmosphere.

Vessel Off-gas Facilities; Stream No. 7 (Fig. 2): The air from the vessel off-gas system is moved by high speed stainless steel exhausters through stainless steel ducts and collected at the central gas disposal facility. Reactive gases, such as iodine, are removed by caustic scrubbers in the processing buildings before the air enters the ducts. The air is cleaned by passing it through a Cottrell electrostatic precipitator followed by a filter identical in performance with those described below for the Graphite Reactor. This system has a capacity of 2000 cubic feet of air per minute and collects gas from all radioisotope production vessels, off-gas from hoods, and ventilation air from dry boxes. The cleaned air is discharged to the atmosphere through a 250 ft stack. A proposed cleaning system to handle an expected increase of 4000 cubic feet per minute of contaminated air is in a preliminary design stage. This new system will handle highly radioactive gases from the Oak Ridge Research Reactor (ORR) experiments as well as from future expansions of the chemical processing Hot Pilot Plant. A caustic scrubber plus high efficiency filters will probably be the cleaning equipment used.

The Cottrell precipitator is of the exposed-tube type, containing twenty-three 8-in.-dia tubes 12 ft in length, fabricated of No. 14 gauge stainless steel. Those parts of the precipitator that come in contact with corrosive gases to be cleaned are constructed of type 347 stainless steel. The discharge electrode system is stainless steel wire longitudinally centered through each collecting electrode, suspended from porcelain insulators and held taut by porcelain weights at the bottom.

The precipitator is equipped with a continuous water-flush system. Nozzles are provided at the top of each tube in such a manner that a continuous film of water is maintained on the inner surface of the tube. In addition, a water-spray system is located at the extreme top to facilitate washdowns. The water is collected in a catch tank and recirculated to the precipitator. When the activity level or the acid content (NO and NO_2 are present in the off-gas) becomes high,

the water is discharged to a monitoring tank in the highly radioactive liquid chemical waste system.

The precipitator is designed to operate at a maximum temperature of 200°F and a negative pressure of 60 inches water at a flow of 2000 cubic feet per minute. The water flush system operates at 3 to 6 gallons per minute per tube.

The electrical design voltage is 75,000 volts, 25 kilovolt-amperes and operates on 440-volts 3-phase 60-cycle current to the power pack. In actual practice the precipitator operates on 52,000 volts and a current of 130 milli-amperes.

Filtered Cell Ventilation Facilities; Stream No. 6 (Fig. 2): Ventilation air from radiochemical processing areas is collected at the central gas disposal facility through steel and concrete ducts. The air is moved through the ducts by high capacity, low vacuum exhausters. The cell ventilation cleaning facility consists of a bank of Fiberglas filters backed by high efficiency paper filters and is essentially a miniature of the Graphite Reactor filter house described below. An additional filter bank, similar in filtering detail, has just been constructed in another area to filter cell air from the Hot Pilot Plant. All cell ventilation air is dispersed by a 200-ft-stack.

Unfiltered Cell and Hood Ventilation Facilities; Stream No. 6 (Fig. 2): A system of steel and concrete ducts collects ventilation air from laboratory hoods and experimental cells in which the possibility of air contamination is slight. This air is dispersed to the atmosphere via a 200-foot stack without preliminary cleaning. High capacity, low vacuum exhausters move the air through the ducts.

Figure 8 is a photograph of the central gas disposal area. To the right behind the concrete barricade is the filter bank for the cell ventilation system. In the left foreground are the main exhaust fans. Figure 9 is a photograph of the same area showing the top of the Cottrell precipitator and the stainless steel discharge line to the stack. Figure 10 is a photograph of the precipitator during construction.

In addition to the main central facility, several small air-cleaning filters are located at isolated experimental radiochemical installations. Except for one unit in which a graded filter-fiber unit developed at Hanford is used, these are minature replicas of the Graphite Reactor filter installation.

ORNL Graphite Reactor Cooling Air; Stream No. 5 (Fig. 2): For cleaning the cooling air from the nuclear reactor, consideration was given to several methods including cyclone separators and electrostatic precipitators. Filtration was selected as the method to be used after extensive literature research and consultation with specialists in air cleaning problems. The short-lived radioactive gases are not removed from the air before discharge to the atmosphere. The filter house is a large reinforced concrete structure composed of four identical cells, each containing a roughing filter and a polishing filter. The air from the reactor enters the top of the filter house, passes downward through the roughing filter, then horizontally through the polishing filter into the exit air duct. Two fans, operating in parallel at a negative pressure of 50 in. water gauge deliver 60,000 cubic feet per minute each of air to a 200-ft stack. The expected

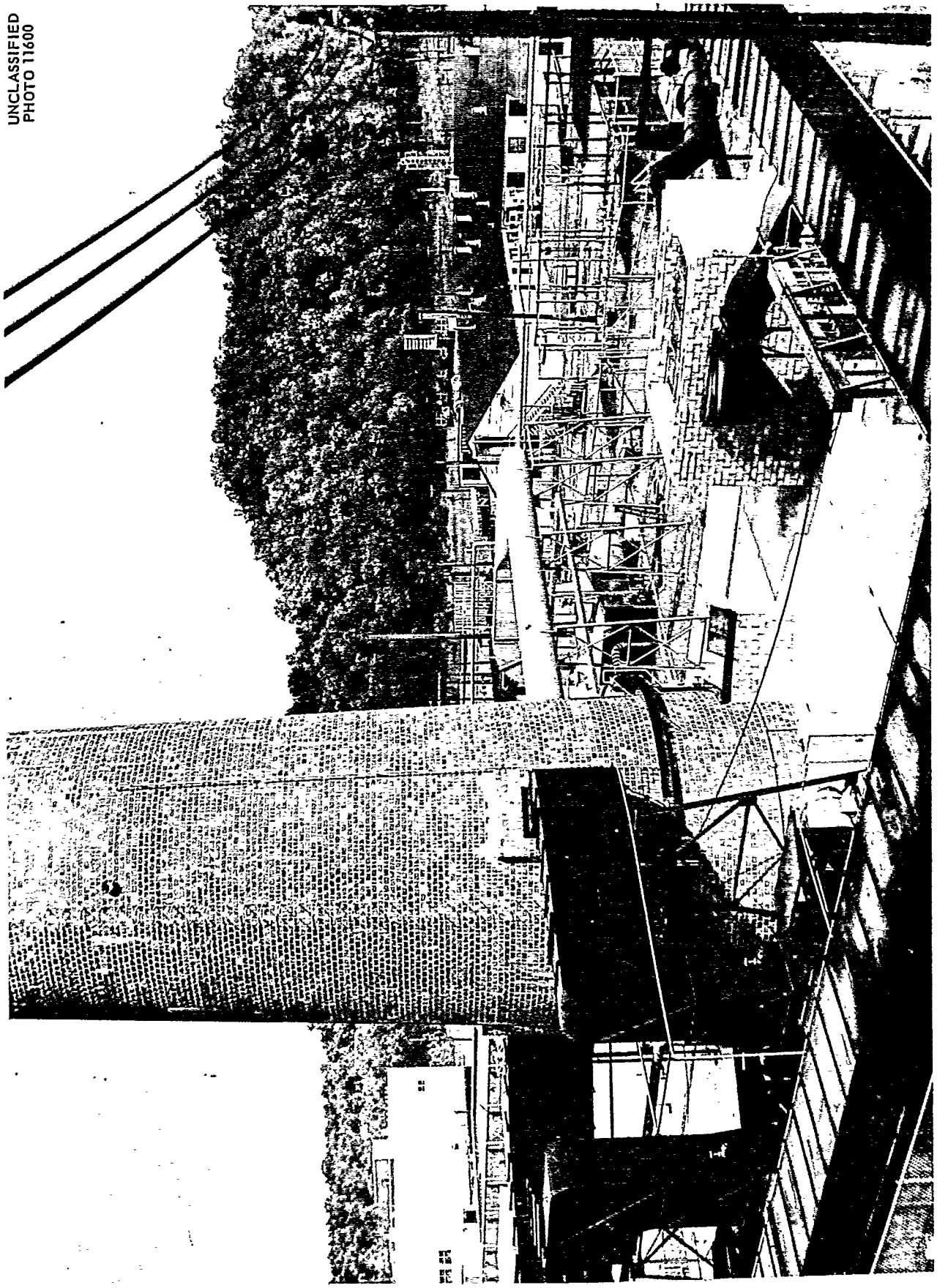

Fig. 8. Central Gas Disposal Area.

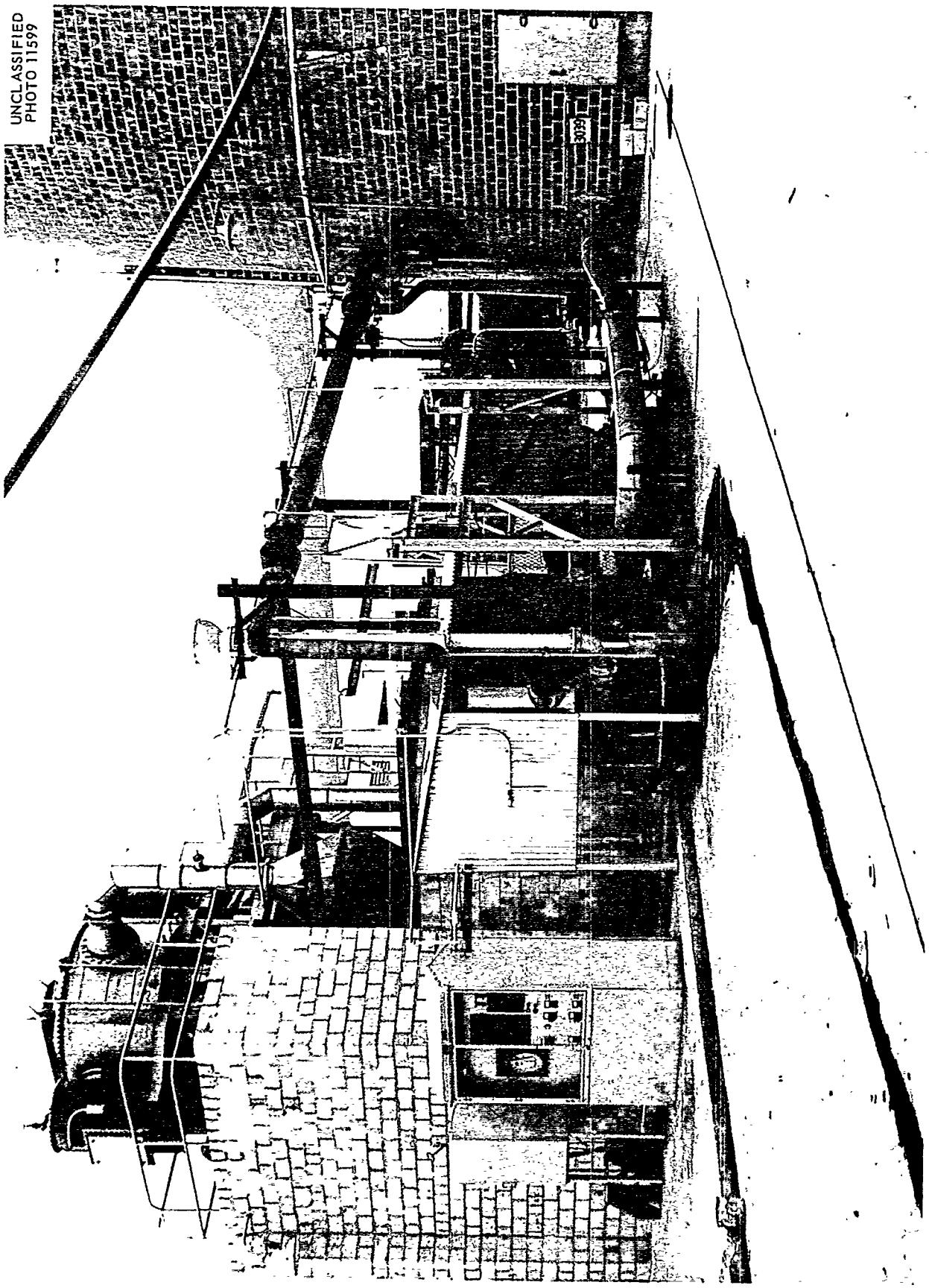

Fig. 9. Central Gas Disposal Area.

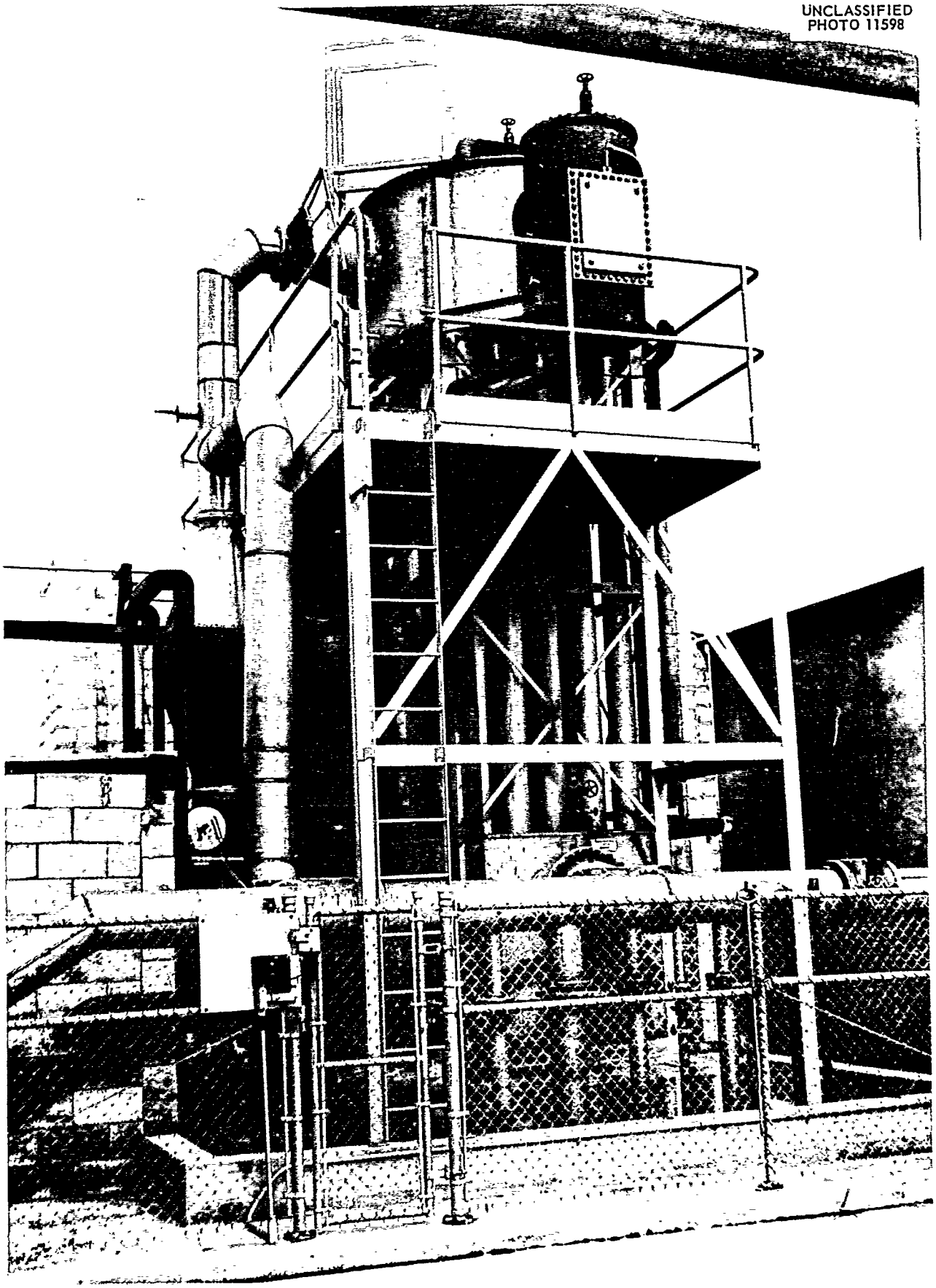

Fig. 10. Central Gas Disposal Area.

dust load was less than 900 grams per day of particle with a maximum diameter of 600 microns, a large number of them being in the submicron range. The designed efficiency of this house exceeded 99.9 per cent for particles down to 0.3 micron in size.

The roughing filters are standard commercial deep-pocket filters, each pocket containing two layers of 1/2-in.-thick graded submicron glass fibers packed to a density of 0.6 pound per cubic foot. The polishing filters are high efficiency paper units developed for the AEC. These filters have a particle removal efficiency of 99.5% at 0.3 micron particle size.

All access to the filters is through removable roof slabs which provide a method for remote maintenance. A canal located across the front of the filter house provides a receptacle for the dust-laden filters when the filter medium is being renewed. Precautions are taken to seal all filters in place in structural steel frames to ensure against leaks and bypassing of the filters.

Maintenance of the filter house is practically nonexistent except for periodic renewal of the filter media. The average life of the roughing filters is two years and of the polishing filters two and one-half years. The roughing filters are changed, one cell at a time, at approximately six-month intervals when the pressure drop across the filter house approaches or exceeds 8 inches water gauge. The filters are washed down, removed, and stored in the canal. The filter media are removed from the pockets and buried in the solid waste disposal area, and the pockets are reloaded for the next change.

The polishing filters are all changed at the same time when the pressure drop across them approaches 5 inches water gauge. To confine the dust, both sides of the filters are sprayed with strip coating before they are removed from the building for burial.

Homogeneous Reactor Test (HRT) Reactor Off-gas: The off-gas from the HRT is a 1/2 liter per minute stream of oxygen saturated with water vapor and containing small quantities of xenon and krypton isotopes. The water vapor is removed by cooling the gas to -30°F. The xenon and krypton isotopes are adsorbed and held on a bed of activated carbon until they have decayed to a very low activity level. The oxygen carrier gas is discharged to the stack.

Four carbon beds are installed in parallel at the HRT. Only two of these beds are expected to be used at any given time; one bed is a spare and the fourth bed serves the processing plant. Each bed consists of 40 ft of 1/2-in. pipe, 40 ft of 1-in. pipe, 40 ft of 2-in. pipe, and 60 ft of 6-in. pipe. The pipes for each bed are filled with about 410 pounds of activated charcoal pellets. The beds are submerged in water (for cooling) in an underground pit.

The off-gas from the reactor passes first through cold traps which freeze out the excess water vapor. The dry gas then passes through a long, empty three-inch pipe. The transit time through this pipe is about 7 hr, long enough for the most active isotopes to decay. The gas then goes to the carbon beds where the xenon and krypton are adsorbed on the carbon and held long enough (about 13 days for the krypton and 210 days for the xenon) for all but the krypton-85 activity

to decay away. The activity discharged to the stack is expected to be about 6 curies per day of krypton-85.

Oak Ridge Research Reactor (ORR) Off-gas: Off-gas from experiments connected with the ORR is discharged into the Laboratory vessel off-gas system, where it is cleaned before discharge to the atmosphere. This cleaning system has been described above.

The ORR building is equipped with a special air removal system for use in the event a catastrophe results in a release of fission products within the building. On activation by a monitoring system located within the ORR building, all doors, windows, etc., are closed; the normal building ventilation system is shut down; and a 5000 cfm emergency exhaust system is started. This emergency exhaust system is designed to maintain the ORR building under a negative pressure of 0.3 in. of water as long as required. This negative pressure ensures that all leakage of air through the building walls will be inward and will be exhausted under controlled conditions for cleaning prior to atmospheric discharge via a 250-ft stack located at the central gas disposal area. Cleaning of the exhaust air is accomplished by filtration for particle removal and scrubbing with caustic solution in a packed tower for iodine removal. Upon activation of the exhaust system, the caustic scrubbing system is started automatically.

IV. Current Handling and Disposal Methods for Solid Wastes; Stream No. 4 (Fig. 2)[11b]

Oak Ridge National Laboratory not only produces quantities of radioactive solid wastes but also serves as the major disposal site for such wastes from Argonne National Laboratory, Knolls Atomic Power Laboratory, Mound Laboratories, Battelle Memorial Institute, General Electric Company in Evandale, Ohio, and other off-site installations. Radioactive solid wastes from these sources occupy approximately half the burial ground space currently being used at ORNL.

Nearly all the solid wastes routinely produced at ORNL consist of trash such as paper towels, glassware and other expendable items contaminated with radioactive materials. Occasionally contaminated concrete and wood and large pieces of equipment that cannot be economically salvaged by decontamination methods must be buried. Compared to the liquid wastes previously described, the solid wastes usually contain minor quantities of radioactivity, probably about 1% of the total disposed of by the Laboratory. Accurate determination of the curies handled is impossible, because the wastes are a nonuniform mixture of many kinds of materials.

In collecting contaminated solid wastes the prime considerations are personnel safety and preventing the spread of contamination. Ordinary galvanized garbage cans are used inside buildings as solid waste collectors. These "hot cans" are painted a distinctive yellow color and marked with a radiation symbol to distinguish them from waste cans used for uncontaminated trash. Hot cans are leak-proof and fitted with suitable covers to prevent the spread of contamination during handling. Frequent routine monitoring of the waste can areas is conducted to be sure that accumulation of materials does not produce radiation levels that will be hazardous. Special leak-proof Dempster Dumpsters are used to collect larger volume wastes. Placed at convenient locations outside certain buildings these are also distinctively marked for contaminated trash collection only.

Personnel who handle radioactive solid wastes are provided with protective clothing--cover-alls, shoes or boots and cloth or rubber covered gloves--to prevent contamination of personal clothing and body parts. Exposure above prescribed limits requires assignment to other work until the weekly average exposure falls below maximum permissible limits. Where an inhalation hazard exists, personnel are provided with filter-type Assault Masks. A radiation survey is made before moving materials to determine the radiation levels present and to establish working time for prevention of personnel exposure in excess of maximum permissible limits. Where radiation levels are sufficiently high to preclude handling the materials manually, remote handling by cranes or behind shields is practiced. An example of this type handling is the removal of "hot" dirt resulting from a liquid waste spill or break in a liquid waste line to the tank farm. All personnel are monitored with film badges and dosimeters to measure their radiation exposure.

Handling of wastes from outside agencies presents a number of problems and requires the use of a varied number and type of personnel. At one time the outside agencies were not required to identify the composition of materials shipped to ORNL as waste. However, after several incidents in handling unknown hazardous chemicals, the shippers have been required to describe their solid wastes in writing before they are accepted. Materials shipped by commercial

carrier must be packaged in such a way that they can be easily handled and so shielded that they will meet U. S. Interstate Commerce Commission regulations for shipment of radioactive materials. Wastes shipped in small lots by truck may require only one or two laborers to unload the materials and a health physicist to monitor the operation. These wastes are dumped directly from the truck into the burial pit.

Large shipments which arrive by rail are surveyed to determine the radiation levels involved before unloading and transfer to the burial ground. If broken containers are found, necessary precautions must be taken to prevent the spread of contamination during unloading and transit to the burial site.

After completion of the burial operation, men, equipment and vehicles are checked and decontaminated as necessary. This may even necessitate tearing out and replacing sections of the floor or other parts of a vehicle. Vehicles cannot be released to a public carrier for commercial transport of other materials until satisfactorily decontaminated.

Waste solids at ORNL are normally transported to the burial site by a group of special trucks used only for this purpose. These trucks are distinctively marked and fitted with leak-proof beds to prevent the spread of contamination during transit. A lead shield is placed between the can and the bed of the truck to provide protection for the crew against direct radiation. The pickup and loading of the materials are monitored by Health Physics personnel and the crew is advised as to the radiation levels involved and the maximum transit time allowable to prevent undue exposure.

Especially hot materials require special precautions. This may consist only of the addition of more shielding to the "hot" truck, but in some cases a crane is used to reduce the radiation hazard to the transport crew.

Radioactive solid wastes are currently buried in a Conasauga shale formation in the waste disposal area of Melton Valley near the highly active liquid chemical waste disposal pits (Fig. 5). Where weathered, the shale is soft and lends itself to easy excavation with power shovels or similar equipment. Normally the burial site is staffed during the work day with only a heavy equipment operator who digs and backfills the burial pits.

Alpha contaminated wastes and beta-gamma contaminated wastes are handled and buried separately. Alpha contaminated wastes are buried in holes approximately 15 by 15 feet deep. The waste materials are placed in the hole and covered by about a foot of earth. Approximately 8 inches of concrete is then poured over the earth in the hole. The concrete is followed by an additional two or more feet of earth to fill the hole to surface level. The use of concrete is a precaution to prevent anyone from inadvertently digging into the long-lived plutonium-contaminated alpha wastes at some future date.

Waste materials contaminated by beta-gamma activity from fission products are buried in trenches approximately 10 feet wide by 15 feet deep, which are dug completely across the burial site. Beginning at one end the wastes are dumped into the trench. When the first portion of the trench is filled, the wastes are

covered by three or more feet of earth depending on the activity of the wastes. This dumping and back-filling procedure continues toward the other end of the trench until it is completely filled.

In special cases where very small, extremely active waste materials must be disposed of, auger holes approximately 1 foot in diameter and 15 feet deep are drilled in the burial site and the materials dropped into the hole. This procedure is used primarily to limit the radiation exposure to personnel during the back-filling operation.

The burial site is laid off in grid sections to facilitate efficient use of the entire area. Records are kept of the type and quantity of materials buried, and their locations are marked on the plot map.

Little information is available relative to the contamination of ground water by leaching or other means. However, the location of the site is such that the direction of ground water flow is toward White Oak Creek and the Clinch River into which low-level liquid wastes have been released almost since the beginning of the project. The burial site is completely fenced to prevent unauthorized entry and marked with radiation signs to warn personnel of the potential hazard.

Burial has been used at Oak Ridge National Laboratory as a means of disposal for solid wastes since the beginning of operations. To date four burial sites have been used. The earlier burial sites were relatively small and located without benefit of geological exploration. In two cases surface erosion was a problem, since the sites were located on the side of a hill. As each site was filled, it was abandoned. The site used prior to the present area was the largest of the earlier sites, comprising approximately 7 acres. It was used for a little over 5 years before being filled at an average annual burial rate of 1-1/2 acres. The current burial rate is of the order of 5 acres per year. This is an increase in the rate of burial site consumption by a factor of 3 in the last ten years. The increase may be attributed primarily to the increased use of ORNL burial facilities by outside agencies for the disposal of their solid wastes. At present, there are 25 or more outside agencies using these facilities. The present burial site is a relatively flat, cleared area of approximately 30 acres, with additional wooded land adjacent that can be used for expansion in future years.[16]

V. Surveys and Studies of Waste Disposal Effectiveness

The release of waste materials, either gaseous or liquid, to the environment places certain responsibilities upon the releasing agency. The releasing agency must safeguard the health and well-being of its employees; it must protect the public health and property in both the immediate and long-term aspects; and finally, it must assume responsibility to its own management with regard to legal implications associated with waste releases. In order to discharge such responsibilities, the Laboratory must necessarily control and evaluate its waste releases. The ORNL Health Physics Division performs this duty by maintaining, within prescribed limits, concentrations of radioactive wastes released by the Laboratory and by monitoring the surrounding area to prevent hazardous conditions that might arise from accumulations or reconcentrations of radioactive material. Several systems are employed to monitor for air-borne contamination and water contamination.

A. Monitoring for Air-borne Contamination

Atmospheric contamination and fall-out in the general environment are monitored by the following methods: (1) the three off-gas stacks in the X-10 area are monitored continuously for the purpose of preventing excessive discharges of radioactivity to the atmosphere for extended periods; (2) samples are collected at a number of stations by passing air continuously through filter paper from which air contamination may be evaluated; (3) fall-out is monitored by means of gum paper fall-out trays located at all sampling stations; (4) background measurements are made at many locations to determine the gradual increase in background radiation and to detect daily fluctuations in background. In addition, fall-out is monitored by collecting rain water samples.

Each air sampling method yields slightly different information relative to the nature of the air-borne contamination. Since the filter sampling equipment is located inside louvered enclosures, the heavier particles settle to the ground without being collected on the filters. This type sample contains only particles which might be considered of breathable size. The fall-out trays collect the heavier particles as well as the light particles and thus present the total fall-out picture. The background measurements indicate the radiation from contaminated soil and from radioactive gas such as natural radon or reactor-produced argon-41. Rain water samples contain both fall-out and "rain-out" and give information on the soluble and the insoluble fractions of the radioactive contamination.

1. Stack Monitoring

Continuous stack monitors are installed on the stacks at the Isotope Area, Hot Pilot Plant, and Graphite Reactor. The equipment at the isotope stack consists of a conventional LaPine motor blower unit which draws air from the stack gas stream through a millipore filter. At the Hot Pilot Plant stack the sample is drawn from the stack stream through a millipore filter by means of a steam jet arrangement. The sample at the Graphite Reactor is drawn from the duct downstream from the filter house and uses the pressure differential across the fan to draw the sample through Hollingsworth-Vose (H. V.) 70 filter paper. Mounted in a shielded box just below the H. V. 70 filter paper is a Geiger-Muller (G. M.) tube which monitors the activity collected on the filter. The counting rate is recorded on an Esterline Angus recorder.

The filters are changed daily at each stack and evaluated for gross beta and gross alpha activity. The data presented below represent samples covering a minimum of one month following installation. Average concentration values for gross beta and gross alpha activity and the highest average value for any 24-hour period from each stack are presented.

Concentration of Radioactivity in Stack Effluents

	Stacks		
	Hot Pilot Plant	Isotope Area	Graphite Reactor
Gross Beta			
Highest average per 24 hours	1.1×10^{-8} μc/cc	9.1×10^{-9} μc/cc	9.5×10^{-8} μc/cc
Average for period	3.2×10^{-9} μc/cc	9.3×10^{-10} μc/cc	3.9×10^{-9} μc/cc
Gross Alpha			
Highest average per 24 hours	1.5×10^{-10} μc/cc	5.5×10^{-11} μc/cc	1.3×10^{-12} μc/cc
Average for period	2.4×10^{-11} μc/cc	2.6×10^{-11} μc/cc	3.4×10^{-14} μc/cc

The average values listed in the table may be considered probable normal operating levels. When concentrations of activity in the stacks exceed these values, the values found must meet the test set forth by the following expression:

$$3\left(\frac{C_b \times 10^{-2}}{MPC_b} + \frac{C_a \times 10^{-2}}{MPC_a}\right) \leq 1$$

C_a = gross alpha concentration in stack effluent

C_b = gross beta concentration in stack effluent

MPC_b = breathing zone MPC (maximum permissible concentration) for gross beta activity

MPC_a = breathing zone MPC for gross alpha activity

The maximum ground concentration per unit discharge rate from each ORNL stack for varying weather conditions has been calculated by Myers and Purdy on the basis of Sutton's equations.[17] The data indicate a probable minimum dilution

factor of the order of 10^3 for each stack. Since a large segment of the Laboratory population will be involved in air-borne contamination from stack discharge, any uncontrolled exposure from this source should not exceed one-tenth the maximum permissible concentration (MPC). Consequently, the dilution factor of 10^3 has been reduced by a factor of ten with the result that 10^{-2} rather than 10^{-3} appears in the above equation. The factor of 3 is applied because there are three stacks which are monitored and evaluated separately. The factor of 3 in the equation obviates the possibility of each stack continuously discharging sufficient activity to create concentration of one-tenth MPC.

It should be emphasized that the above equation was developed to determine an action point for field investigation and subsequent corrective measures. If the left-hand side of the expression is greater than one, but less than ten, the condition is investigated. Identification of specific isotopes and consideration of their particular MPC values may reduce the quantity of the left-hand side to an acceptable value. If this is not the case, remedial action is taken. Should the value of the left-hand side exceed ten, remedial action is taken immediately and followed by an investigation.

2. Air Sampling and Fall-out Collection

a. Filter-type Air Monitors. Air samples are collected at all stations (local, perimeter, and remote) by passing air continuously through H. V. 70 filter paper. At the local stations the activity collected on the filter is continuously monitored by means of a Geiger-Muller tube and logrithmic count rate meter, the readings being recorded on a Speedomax recorder (Fig. 11). In addition, the counting rates are telemetered via telephone lines to the area monitoring control laboratory where they are recorded on a twelve-point Speedomax recorder; each station reading is printed once every twelve minutes. The one-minute interval between station printings is necessary to compensate for the slow time constant inherent in the logarithmic rate meter circuit. The perimeter and remote stations contain no monitoring and recording equipment (Fig. 12). Under normal conditions of air contamination, the filters are removed once each week. When air contamination is significant, more frequent changes are necessary. The charts from the recorders in the local stations are changed normally at intervals of approximately three days. An hourly tabulation is made of the counting rate of the activity on each filter. Where air contamination in significant quantities is encountered and the half-life of the activity is long in comparison with the collection time, an indication of the concentration of activity in the air can be obtained from the rate of increase in activity on the filter. After removal, all filters are brought to the monitoring laboratory for analysis.

b. Gummed Paper Fall-out Trays. Radioactive fall-out is sampled by means of sedimentation trays or frames, each consisting of an aluminum frame (inside demensions 12 in. x 12 in.) to which is affixed a cellophane sheet gummed on one side (Fig. 13). The frame with the gummed paper is placed on a horizontal shelf attached to the air monitoring station. The gummed surface is exposed in an upward direction so that any particulate matter which falls upon this gummed surface becomes firmly fixed.

Fall-out samples are collected at all stations. Duplicate sets of fall-out frames are maintained; while one set is in the field for collection of particles,

Fig. 11. ORNL Local Air Monitoring Station.

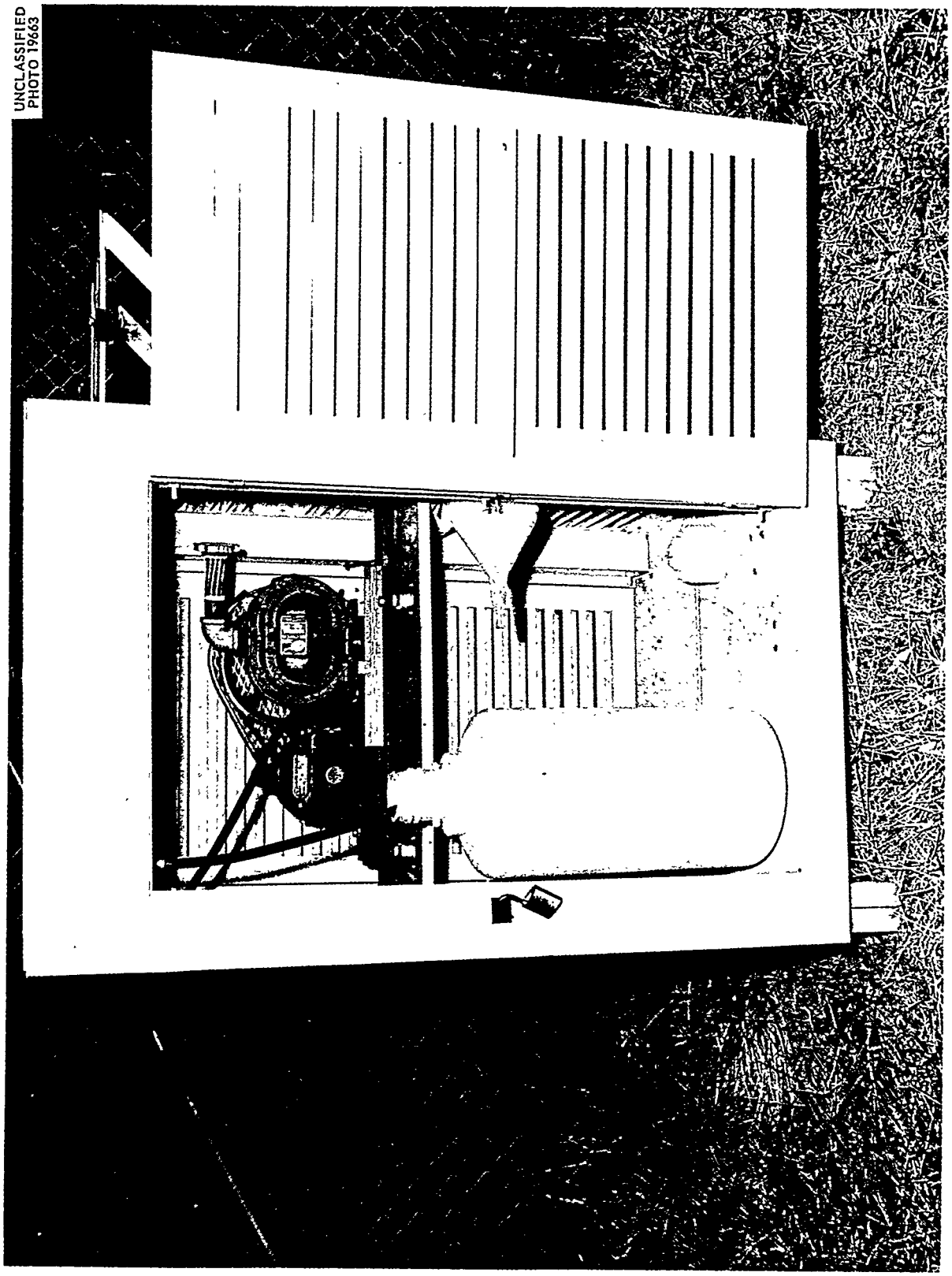

Fig. 12. ORNL Perimeter and Remote Air Monitoring Station.

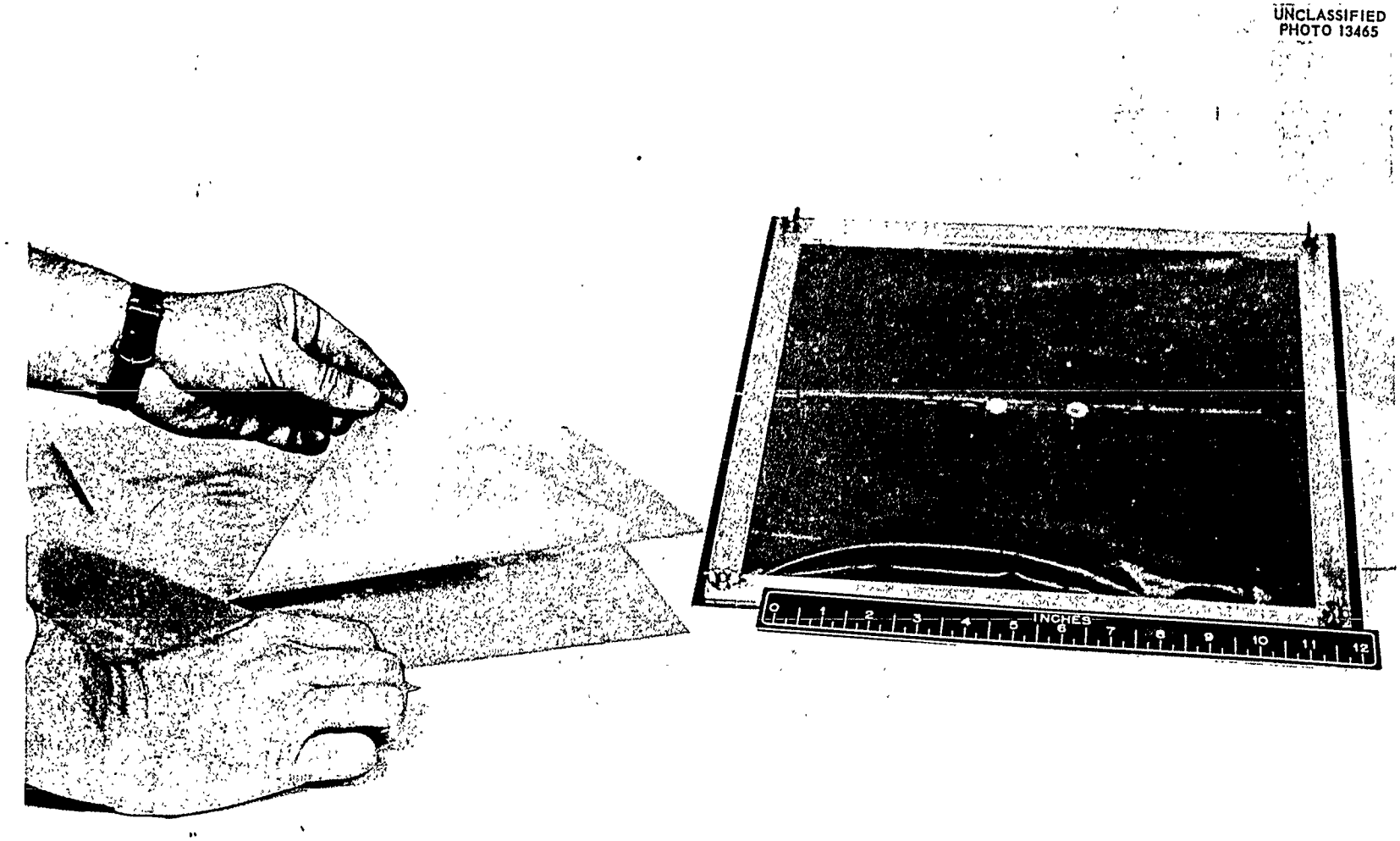

Fig. 13. ORNL Fall-out Sedimentation Tray.

the other set is in the laboratory for processing and analysis. Frames are exchanged normally once each week and transported in a special box which keeps them separated during transit.

c. Rain-out Samples. Rain-out measurements are made by collecting rainfall in 30-in. x 30-in. trays located at all perimeter and remote stations and one of the local stations. The collection tray serves as the roof of the monitoring station (Fig. 12). All rain collected is drained from the bottom of the tray into a polyethylene container inside the station. The depth of water in the container is measured to determine total rainfall from a calibration curve. Samples are collected normally once each week. The water in the container is shaken and one liter of the homogeneous mixture is taken to the monitoring laboratory for processing. The remainder of the water is discarded. The collection trays and the containers are cleaned at periodic intervals to prevent buildup of contamination from previous rains.

d. Analytical Methods. In the laboratory continuous air monitoring filters are mounted on 14-in. x 17-in. cardboards for autoradiographing. Twelve filters are mounted on each cardboard with the collecting surface of the filter facing away from the cardboard. The mounted filters are placed in the dark room in direct contact with a sheet of 14-in. x 17-in. Kodak Blue Brand X-ray film for an exposure time of 24 hours. The film and filters are weighted to assure complete contact during exposure.

The gummed papers are covered with a sheet of rubber hydrochloride, cut from the fall-out frame, and mounted on a 14-in. x 17-in. cardboard for autoradiographing, which follows the technique outlined above for continuous air monitoring filters.

With the use of a light table and the unaided eye, films are analyzed for radio-particulates by counting the number of particle images appearing on the films. In addition to counting the number of particles, some measure of each particle's activity is obtained by classifying each particle according to image diameter into one of several particular size ranges. The classification is accomplished by comparing image sizes with calibrated image sizes prepared from known particles. Four arbitrary order of magnitude size ranges are: $< 10^5$ d/24 hr,* $10^5 - 10^6$ d/24 hr, $10^6 - 10^7$ d/24 hr, $> 10^7$ d/24 hr. The results are tabulated for study and comparison with previous data and norms.

Following autoradiography, the continuous air monitoring filters are counted for gross beta activity to determine the average concentration of radioactivity in the air sampled. To evaluate the activity on the sample, the filter is affixed to the inside wall of a 1-in.-diameter aluminum cylinder, which is placed in a counter having a G. M. tube positioned along the axis of the cylinder. The sample is counted to determine the total activity in terms of μc/sample. The average air concentration in μc/cc is determined by dividing the total activity per sample by the volume of air sampled.

The gummed papers are each folded into a small square, placed in a crucible, and ashed in a muffle furnace. The ash is transferred to a stainless steel counting dish using a technique of transfer as follows:

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* d = disintegrations.

1. Spray the bottom of the counting dish with a wet film of acrylic resin.

2. Transfer the ash to the counting dish by sprinkling the ash onto the wet resin coating. Use a spatula to disperse the ash evenly over the dish.

3. Dry under a heat lamp. The resin bond, when thoroughly dry, provides a secure bond for retaining the ash.

After drying, the samples are counted in a gas-flow proportional counter to determine the gross beta activity. From these counts, the fall-out of gross beta activity in terms of μc/sq ft is determined.

Rain water samples are vacuum-filtered through Whatman 42 filter paper to remove the insoluble fraction of the activity. The filtrate is placed in a 1500-ml beaker and concentrated by evaporation on a hot plate to approximately 50 ml. The concentrate is then transferred to a 100-ml beaker (using nitric acid and distilled water) and evaporated to a few ml. This is transferred to a stainless steel counting dish and evaporated to dryness under a heat lamp. Care is taken to scrub down the walls of the beakers during each transfer. Each filter, containing the residue or insoluble fraction of the activity, is placed in a crucible and ashed in a muffle furnace. The ash is then transferred to a stainless steel counting dish using the method described for transfer of gummed paper ash.

All samples of both the soluble and insoluble fractions are counted in a gas-flow-type proportional counter to determine the gross beta activity. From these results, the concentration of activity in rain-out in terms of μc/cc is calculated.

When significantly high activity is encountered, an attempt is made to identify the specific radioisotopes and to determine their probable source. The samples are subjected to gamma spectrometry measurements and radiochemical analysis by the Analytical Chemistry Division to identify the radioactive constituents. In addition, the effective half-life and distribution of products in the sample are studied to differentiate between ORNL-originated activity and possible contamination from weapons tests.

e. Evaluation and Effectiveness of Air Sampling. From the air monitoring samples it is possible to determine the general level of air and soil contamination in the area, the source of specific contamination problems, and the improvement achieved over the years by better contamination control. Should the radioactive contamination reach a level that is a significant fraction of allowable limits, investigations into the source of the contamination are made and remedial actions taken immediately. Examples of such instances are the times when uranium oxide particles (and later protactinium particles) were found to be falling out on the X-10 area. Investigations led to the source of the particles, and the processes producing the particles were stopped until corrective measures had been completed. The remedial action required in the case of the uranium oxide particles was additional filtration of the off-gas system and a program for routine scanning of the Graphite Reactor for defective fuel elements. In addition, to prevent the particles already on the area from becoming air-borne and thus increasing the breathing hazard, the roofs and roadways in the area were washed and the barren areas within the X-10 area sown with grass.

An indication of the effectiveness of the air monitoring program for the X-10 area is the fact that the average air contamination value for 1957 was less than 0.1% of the maximum permissible concentration for continuous exposure. The maximum concentration measured was 170% of the maximum permissible concentration. However, this concentration continued for only a short period of time; and, therefore, the concentration averaged over a period of one week did not exceed the maximum permissible for that period.

To determine the air contamination in nearby populated areas, samples have been collected continuously at stations from 10 to 120 miles from ORNL. Although the activity levels have not been considered hazardous, the accelerated programs at ORNL necessitate continuing vigilance. The number and the efficiency of monitoring stations within an approximate 30-mile radius of the Laboratory are being increased.

f. Summary Data from Air Sampling and Fall-out Collection.

(1) Local Area. Average air contamination for 1957 in the X-10 area, determined from measurements made by ten continuous air monitors, was 0.1% of the MPC. The maximum weekly average measured by a single air monitor was 6.0% of the MPC. The maximum concentration measured was 170% of the MPC. In no period did the weekly average of activity concentration exceed the maximum permissible concentration. Table II shows the average and maximum contamination over the last 9 years. The weekly average number of radioactive particles per 1000 cu ft of air sampled was 4.2. The maximum number of radioactive particles for one week was 54.9. Table III shows the average and maximum number of particles over the past 9 years.

Gummed paper fall-out tray measurement gave a weekly average value for 1957 of 4.0×10^{-3} $\mu c/ft^2$. The highest total weekly fall-out for a single collection station was 8.4×10^{-2} $\mu c/ft^2$. The average number of radioactive particles per sq ft per week was 727. The maximum number of particles for any one week was 22,146.

The weighted average value for radioactive contamination in rain water, collected at one station only, was 7.6×10^{-7} $\mu c/cc$. The maximum concentration in any one week was 5.7×10^{-6} $\mu c/cc$.

(2) Immediate Environs (Within AEC Controlled Area). Average air contamination levels for 1957 in the immediate environs, determined by measurements at seven monitoring stations, was 0.02% of the MPC. The maximum weekly average for a single monitor was 2.39% of the MPC. The weekly average number of particles per 1000 cu ft of air sampled was 2.0 and the maximum number for one week was 20.2.

Gummed paper fall-out tray measurements gave a weekly average value of 9.2×10^{-4} $\mu c/ft^2$. The highest total weekly fall-out for a single collection station was 1.4×10^{-2} $\mu c/ft^2$. The average number of radioactive particles per sq ft was 563.9. The maximum number of particles for any one week was 25,848 per sq ft.

TABLE II

Air Activity Measured by Continuous Air Monitors

Units of 10^{-13} µc/cc

Year	Local Area		Perimeter Area		Remote Area	
	Max.	Average	Max.	Average	Max.	Average
1949	2100.0	210.0				
1950	2040.0	180.0				
1951	1050.0	170.0				
1952	782.3	29.0				
1953	4407.3	46.4				
1954	203,183.5	478.4				
1955	2968.6	49.4				
1956	804.7	27.9	29.8	7.0	43.1	10.2
1957	5607.0	82.3	229.7	14.8	47.3	9.0

TABLE III

Particle Concentration Measured by Continuous Air Monitors

Particles Per 1000 cu ft

Year	Local Area		Perimeter Area		Remote Area	
	Max.	Average	Max.	Average	Max.	Average
1949	5.4	0.2				
1950	17.0	0.6				
1951	22.0	0.6				
1952	219.1	3.4				
1953	45.5	2.1				
1954	23.4	1.0			3.3	0.4
1955	13.0	1.4			29.2	1.0
1956	26.6	2.1	8.4	0.7	6.5	0.8
1957	54.9	4.2	16.3	2.0	15.8	1.9

The weighted average value for radioactive contamination in rain water was 8.1×10^{-7} μc/cc. The maximum concentration for one week was 5.1×10^{-6} μc/cc.

(3) Remote Areas (Outside AEC Controlled Area). At present, the Laboratory is operating one monitoring station outside the AEC controlled area approximately 120 miles from ORNL. The average contamination for 1957 measured by this station was 0.01% of MPC. The maximum for one week was 0.05% of MPC. The weekly average number of particles per 1000 cu ft of air sampled was 1.9 and the maximum number for one week was 15.8.

Gummed paper fall-out tray measurements gave a weekly average value of 4.7×10^{-4} μc/ft^2. The highest total weekly fall-out was 2.1×10^{-3} μc/ft^2. The average number of radioactive particles per sq ft was 69.2. The maximum number for one week was 2330.

The weighted average value for radioactive contamination in rain water was 8.1×10^{-7} μc/cc. The maximum for one week was 1.8×10^{-6} μc/cc.

3. Background Measurements

Contamination of the ground in the Laboratory area is monitored by means of background measurements at some 50 locations. The equipment used consists of a calibrated scaler, G. M. tube, and a portable 600-watt Fairbanks-Morse generator power supply. A pick-up truck is used to transport the equipment from station to station. With the equipment on the truck, the G. M. tube is mounted in a fixed position at the end of a 2-in. x 2-in. timber extending horizontally from the bed of the truck. A count is taken at each station and the results converted to mr/hr.*

Measurements indicate that the background level has increased only 0.005 mr/hr in the nearest residential community, Oak Ridge, Tennessee. The average level in the X-10 area during 1957 was 0.12 mr/hr or about 8 times the level 15 years ago (before the operation of the Laboratory); and in terms of present levels of acceptable exposure, this is not considered to represent a hazard. In fact, the background level from natural sources of radiation (e.g., granites, phosphates, etc.) is higher in many sections of the United States than the background in the X-10 area.

B. Liquid Waste Monitoring for Water Contamination

1. Stream Monitoring

a. ORNL Sampling Techniques. At ORNL the major volume of the liquid wastes are discharged via the settling basin to White Oak Creek. The discharge flow is measured by a 90° V-notch weir in conjunction with a standard stilling

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* The roentgen, r, is that quantity of x or gamma-radiation such that the associated corpuscular emission per 0.0001293 gram of air produces in air ions carrying 1 electrostatic unit quantity of electricity of either sign. One milliroentgen (mr) is 0.001 r.

well and Stevens water level or head recorder. The waste stream is sampled by a Trebler Proportional sampler equipped with a revolving dipper so shaped that individual dips collect a volume proportional to the flow through the weir at the time of each dipping. The sampler is equipped with a timer to limit the number of dips per hour, thereby limiting the sample to a volume suitable for handling and analysis.

Grab samples are collected daily at stream stations on Melton Branch and White Oak Creek (Fig. 1). These samples are used primarily to determine whether the activity detected is from the Laboratory proper or from the Melton Valley area (Section II, D). At White Oak Dam, located downstream from the confluence of White Oak Creek and Melton Branch (Fig. 1), continuous samples are collected for daily analysis.

b. Analytical Methods. Four types of samples are processed in the monitoring laboratory. First, those taken from the settling basin and White Oak Creek are evaluated for gamma activity to determine submersion exposure. Second, all of these are checked for gross beta activity. Third, composite samples of both settling basin and White Oak Creek effluent are analyzed radiochemically each week for plutonium content. Fourth, composite samples of both settling basin and White Oak Creek effluent are analyzed radiochemically each month for specific long-lived fission products.

Submersion exposure in terms of mr/hr gamma activity for settling basin and White Oak Creek is determined as follows:

> Samples are put into 14-in.-diameter stainless steel containers to a depth of approximately 12 in. and counted by a calibrated G. M. tube enclosed in a water-tight brass cylinder placed along the axis of the container. The wall thickness of the brass tube around the G. M. tube is sufficient to cut out all beta activity. The counts obtained are corrected for background and the activity in mr/hr calculated.

> The beta activity is determined by evaporating an aliquot of the sample to dryness in an aluminum counting dish and counting in a standard end-window counting set-up. The net counts per minute obtained are converted to 10% geometry and the beta activity in μc/cc calculated. The total curies of beta activity released is calculated from the flow measurements at the settling basin and White Oak Creek and from measurements of the specific concentration of activity in the effluent.

From the daily samples, a weekly and monthly composite is prepared for more detailed analysis for plutonium and long-lived fission products. The weekly composite samples are analyzed for plutonium to evaluate the alpha activity discharged. The monthly composite samples are concentrated and sent to the Analytical Chemistry Division for analysis for long-lived fission products by standard radiochemical procedures. The elements analyzed are strontium, cerium, ruthenium, iodine, zirconium, trivalent rare earths, niobium, barium, and cobalt,

From the analytical results, the fraction of the total beta activity comprised by each long-lived isotope is determined. A weighted MPC value for the mixture is calculated by the formula:

$$MPC_{mix} = \frac{\sum f_i}{\sum \frac{f_i}{MPC_i}}$$

where

f_i = fraction of the total beta each isotope represents

MPC_i = MPC value for each isotope

The calculated concentration in the Clinch River is compared to this weighted MPC value as a measure of the hazard existing.

c. Calculations. The submersion exposure to gamma radiation is calculated by multiplying the net counts per minute obtained with the calibrated G. M. tube by the appropriate conversion factor.[19]

In order to calculate the quantity of beta activity released, a knowledge of the flow and specific concentration from a given source is necessary. The specific concentration is obtained by analysis as previously described. The flow, in the case of the settling basin, is obtained from a continuous recording of head in the weir box and a rating table for the weir. Flow in White Oak Creek is obtained by summing the flows of White Oak Creek and Melton Branch, measured above White Oak Lake basin, and multiplying by a correction factor based on run-off areas to correct for the inflow between the gauging stations and outfall into the Clinch River. The flows in the two creeks are supplied by the U. S. Geological Survey, Surface Waters Branch, and are obtained from steam gauging measurements and continuous recordings of creek stage. The basic formula for determining the specific beta concentration of activity in the effluent is:

$$S_B = \frac{N}{2.22 \times 10^6}$$

where

S_B = gross beta concentration in $\mu c/cc$

N = net d/m/cc

2.22×10^6 = d/m/μc

The formula for the total curies of beta activity discharged is:

$$C_B = \frac{S_B \times F}{10^6}$$

where

F = total flow for the period in cc

10^6 = μc/curie

The concentration of activity in the Clinch River, which is the primary concern, is obtained by multiplying the specific concentration found in White Oak Creek by a dilution factor obtained from a ratio of the White Oak Creek flow to the Clinch River flow at the point of entry. This calculated concentration in the Clinch River assumes intimate mixing with White Oak Creek water upon entry of creek water into the river.

The average daily flow in the Clinch River at the point of entry of White Oak Creek is determined in the following manner. The 24-hour average (midnight to midnight) discharge flow from Norris Dan for the day in question and the preceding day are averaged and subtracted from the 24-hour average (midnight to midnight) Clinch flow obtained at the Scarboro gauge (Cl. 39.0) on the day in question to determine the local inflow below Norris. This figure is multiplied by 435/388, which is the ratio of the drainage area above White Oak Creek (Cl. 20.8) to that above Scarboro gauge to correct for side drainage below Scarboro gauge. To determine the flow at Cl. 20.8 this resulting amount is added to the average Norris discharge previously subtracted.

d. Clinch River Monitoring. Routine surveys of the Clinch and Tennessee Rivers are conducted to determine the extent of dispersion of radioactive material in river sediment, the levels of radiation encountered, and the consequent hazards to humans. Periodic measurements are required to predict the rate of buildup and, consequently, to determine the effectiveness of the liquid waste management program. Also, information is obtained relative to the effect on future industry if the radioactive content of bottom sediment in the Tennessee River systems is increased.

"Cross-section" measurements are taken every two miles in the Clinch River and approximately every 10 miles in the Tennessee River and its reservoirs (Fig. 14) A cross-section study consists of making radiation measurements of the bottom sediment activity and collecting sediment samples at predetermined intervals along the traverse from one bank to the other. Sampling points are located on TVA navigation charts and cross sections taken across the river at these points. Fifty-foot intervals are used in the Clinch River, but an average of ten readings and samples are taken per traverse in the Tennessee River and in Watts Bar and Chickamauga Reservoirs.

A device called a "flounder"[20] is used to measure the gamma radiation of the bottom sediments. The flounder consists of twelve battery-operated G. M. tubes connected in parallel. Pulses from the G. M. tubes are pre-amplified and recorded on a battery-operated scaler, the average count being determined by timing with a stop watch. Samples of bottom sediment for laboratory analysis are obtained with an Eckman dredge.

The gamma measurements made on the bottom sediment are corrected for background and averaged for each cross section. Plots of the average count vs. river mile for the Clinch and Tennessee Rivers are made. Sediment samples are radiochemically analyzed for the long-lived radioactive isotopes present.[21]

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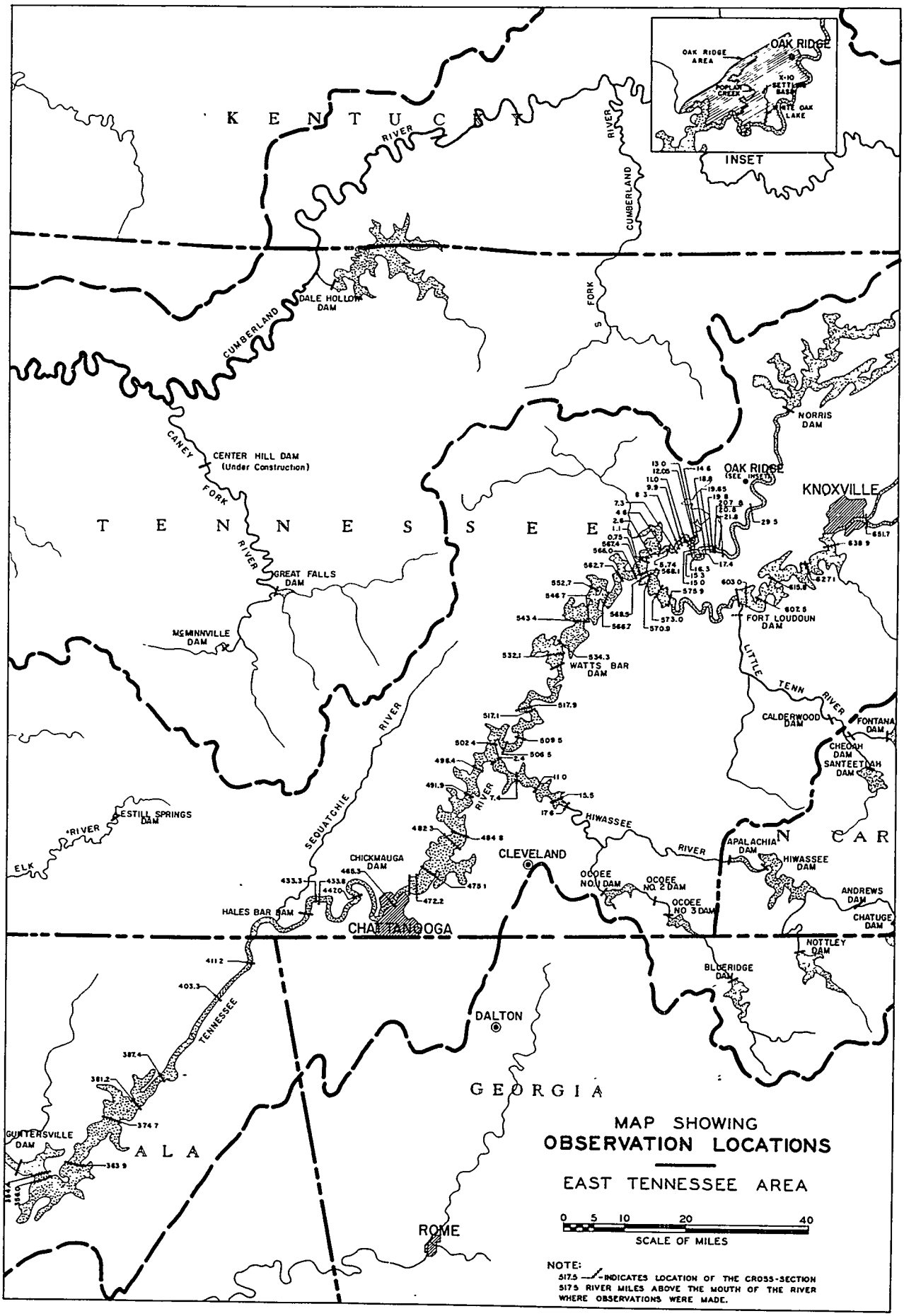

Fig. 14. Tennessee Valley Map of Observation Locations.

A water sample is collected daily at Centers Ferry near Kingston, Tennessee, and composited for a three-month period. The composite is filtered and the filtrate concentrated. The residue or suspended solids and the concentrate are analyzed for fission products to determine the level and composition of the activity existing in the Clinch River.

From time to time, fish from the public waterway are obtained and analyzed for radioactivity. No significant levels of activity have been detected in these fish.[22] Samples of water from the river at the Oak Ridge Gaseous Diffusion Plant and at Kingston, Tennessee, the nearest population centers downstream, have indicated detectable amounts of fission products, but levels are well below those permitted for potable water. In fact, the drinking water for employees at the Gaseous Diffusion Plant is obtained from the river at a point approximately 8 miles below the outlet of White Oak Creek.

e. Summary Data. The total curies of fission products discharged from White Oak Creek to the Clinch River in 1957 was 397. The weekly average gross beta concentration in the Clinch River for 1957 was 0.9×10^{-7} μc/cc. The operating limit of 10^{-7} μc/cc gross beta activity in the Clinch River was exceeded 28% of the time during the year.* Table IV presents data on liquid waste released for the past 10-year period. Table V presents radiochemical analysis data for White Oak Creek effluent for the past 5-year period.

Radiochemical analysis of contaminants in the Clinch River shows activity for 1957 to be the following isotopes: TRE-28.8%, Ce-3.2%, Ru-15.0%, Zr-5.7%, Cs-22.4%, I-0.3%, Sr-20.9%, Cb-1.8%, Ba-0.4%, Co-1.2%.

The concentration of radioactivity in the sediment of the Clinch River drops off materially after the first 20 miles and approaches background levels at 150 miles. The maximum average activity for a single cross section is approximately 18 times background (approximately 0.1 mr/hr) and was found to be located at 12.8 miles below the outfall of White Oak Creek. At 100 miles downstream the level is approximately 2 times background and specific fission products are detectable by silt analysis.

2. Core Hole Monitoring

Evaluation of the effect of liquid waste with regard to the underground water table in the main X-10 area, in the burial ground, and in the area of the waste pit operation is made by means of core holes or monitoring wells. In the main X-10 area, measurements of underground activity detected by core hole monitoring reveal levels to be insignificant. In the waste pit and burial ground area, where most of the disposable high-level waste materials are sent, activity has been detected in the monitoring wells, but it has been established that the contaminant is primarily ruthenium-106, for which the maximum permissible concentration (MPC) in water is quite high, over a hundred times higher than the MPC for strontium-90. Several surface seeps have developed in the pit area, but the location of these areas is such that the direction of ground water flow is toward White Oak Creek and the Clinch River drainage basin into which low-level liquid wastes have been released and monitored since the beginning of the Laboratory.

* By calculation.

TABLE IV

Calculated Values of Radioactive Contamination Discharged to Streams

Year	Settling Basin Beta Curies Discharged		White Oak Creek Beta Curies Discharged		Probable Av. Concentration in Clinch River Units of 10^{-7} µc/cc		Plutonium Concentration Settling Basin Units of 10^{7} µc/cc		Plutonium Concentration White Oak Creek Units of 10^{-9} µc/cc	
	Weekly Av.	Max.	Weekly Av.	Max.	Weekly Av.	Max.	Weekly Av.	Max.	Weekly Av.	Max.
1948	17.3	38.1	9.5	44.3	1.2	8.1				
1949	19.7	5.1	13.8	65.5	2.2	10.4				
1950	3.3	21.1	3.7	16.8	0.5	1.7	1.1	10.0	3.4	12.8
1951	3.3	20.0	2.0	6.8	0.3	1.7	1.1	6.3	4.7	42.1
1952	7.5	35.6	4.1	16.9	0.6	2.5	1.5	10.1	4.2	19.0
1953	5.6	34.8	5.8	27.5	0.9	6.3	1.5	19.7	8.5	112.4
1954	4.5	41.8	7.4	31.0	1.7	9.4	2.0	22.5	6.9	37.0
1955	4.1	16.6	8.4	53.6	1.3	12.7	2.5	36.1	18.6	173.3
1956	4.9	23.3	11.2	66.4	1.5	9.3	1.9	12.6	27.8	151.2
1957	3.6	13.3	7.6	22.4	0.9	3.0	1.0	3.7	12.2	35.5

TABLE V

Yearly Discharges of Radioactivity to Clinch River, 1950-1957

Year	Gross Beta[a] Curies	Per Cent of Gross Beta Activity Identified With Specific Radionuclides[b]											
		TRE[c] (-Ce)	Ce	Ru	Zr	Cs	I	Sr	Nb	Ba	Co	MPCw x 0.10 (Calculated)[d]	Probable Av. of Discharge[e]
1950	191												
1951	101												
1952	214												
1953	304	37.4	2.2	8.7	2.5	2.1	0.7	44.3	1.2	0.9	---		
1954	384	42.6	6.2	2.9	3.6	5.7	0.9	35.2	2.4	0.6	---	3.1×10^{-7}	1.7×10^{-7}
1955	437	33.7	19.4	7.0	1.2	14.4	1.6	21.1	1.3	0.2	1.5	4.4×10^{-7}	1.3×10^{-7}
1956	582	23.8	10.1	5.0	2.0	29.6	0.6	17.9	2.6	0.5	7.9	6.3×10^{-7}	1.5×10^{-7}
1957	397	28.8	3.2	15.0	5.7	22.4	0.3	20.9	1.8	0.4	1.2	5.3×10^{-7}	0.9×10^{-7}

[a] Gross beta activity in daily samples from Creek at White Oak Dam.
[b] Daily samples at Dam composited for monthly radiochemical separations and analysis.
[c] Trivalent rare earths exclusive of cerium.
[d] Maximum permissible concentrations for drinking water are calculated on the basis of the percentage of the specific radionuclides identified. MPC values for occupational exposure are determined and reduced by a factor of ten as recommended by the ICRP for drinking water for the population near a nuclear installation.
[e] Daily volumes of flow are determined from records of USGS gaging stations on White Oak Creek and Clinch River. The probable average concentration in Clinch River, as calculated, is the observed concentration in the Creek multiplied by the dilution factor (i.e. $\frac{\text{Flow in Creek}}{\text{Flow in River}}$) averaged over the same period of time - daily, weekly, monthly, or yearly.

C. Conclusions

In conclusion, Oak Ridge National Laboratory is contributing some radioactive materials to the local environment, and continuing vigilance must be maintained to determine the nature and extent of dispersion of these radioactive materials. Up to the present, data collected from the current monitoring systems indicate that the activity released is below the maximum permissible levels recommended by the National Committee on Radiation Protection and the International Commission on Radiological Protection. The low degree of radioactive contamination of the air and water by ORNL does not represent a hazard to the local environment or population.

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